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MATERIALS INTEGRITY SECTION
A REVIEW OF 1987 AND PROPOSALS FOR 1988

Report No 88-35-K

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ABSTRACT

This report describes the work programs performed in 1987 by the Materials Integrity Section of the Metallurgical Research Department. The direction of the 1988 work programs is outlined.

job	file	date	report no.
84-875-0200	823.1	March 22, 1988	88-35-K

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EXECUTIVE SUMMARY

MATERIALS INTEGRITY SECTION: A REVIEW OF 1987 AND PROPOSALS FOR 1988

R.G. Fleck

The Materials Integrity Section is extensively involved in the areas of delayed hydride cracking and fracture of pressure tubes and the effects of irradiation on the deformation of current and modified pressure tubes. From these studies the Materials Integrity Section provides to Ontario Hydro an in-depth knowledge of the performance of zirconium alloy components in the core of CANDU reactors. In support of thermal plant a corrosion-fatigue test facility for boiler tube materials has been developed.

Several key developments occurred in 1987. A technique for the non-destructive measurement of hydrogen in pressure tube materials was developed. Also based on experience and a test program, it was demonstrated that hydrogen diffused away from fret marks in pressure tubes. This result contributed to an AECB decision to waive removal of a pressure tube from Pickering 5 to determine the hydrogen distribution at fret marks which were developed in the hot commissioning of this unit. The development of a probabilistic model to predict the number of blistered tubes in reactors where pressure/calandria tube contact occurs has provided a basis to assess the condition of the core of Pickering Units 3 and 4 after the higher than anticipated deuterium levels were detected in a P3 pressure tube. The probabilistic model was developed based on the expertise developed, within the Section, on blister growth and fracture. Important input to the model was the prediction of the position and rate of increase in the contact area between the pressure tube and the calandria tube. This prediction was made based on an understanding of irradiation enhanced deformation and on the applicability of the design equation to determine the sag of the pressure tube.

Some irradiation growth specimens in the Dido reactor have now reached fast neutron fluence levels which exceed the average fluence levels in central pressure tubes in the lead reactor. Specimens prepared from surveillance pressure tubes from Pickering 4 and Bruce 2 have been included in the Dido study. Transmission electron microscopy has also been performed on material from these pressure tubes and similar work is planned for specimens that have been irradiated in Dido.

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TO: D. Mills
Director of Research

MATERIALS INTEGRITY SECTION
A REVIEW OF 1987 AND PROPOSALS FOR 1988

1.0 INTRODUCTION

Throughout 1987, the Materials Integrity Section has made significant contributions to the Ontario Hydro CANDU program. From studies of hydrogen diffusion, it was possible for Ontario Hydro to demonstrate to the Atomic Energy Control Board that a fretted tube from Pickering 8 did not require to be removed for examination. Based on a probabilistic model, which was initiated under OHRD OM&A funds, assessment of the condition of the P3 and P4 pressure tubes has been made. This assessment was required after higher than anticipated levels deuterium were detected in a P3 pressure tube (LO9). The results indicate that each reactor could be run for an additional 7000 EFPH before critical blisters can be expected. The development of the predictive model also required input from deformation aspects. Of importance, was to establish the rate of increase in the contact zone as the pressure tube sagged down onto its calandria tube. To develop such an assessment required an understanding of deformation and in particular on how the design equation can be applied to predict pressure tube sag. Such an understanding is available and is a result of long-term studies of in-reactor deformation.

The above responses to immediate needs of the OH CANDU program make it difficult for the limited resources of the Materials Integrity Section to meet schedules imposed by the CANDU Owners Group (COG). The programs supported by funding from COG are generally aimed at longer-term concerns with regard to fuel channel performance in CANDU reactors. The financial support from COG is required to provide sufficient funds to maintain the staff and programs of the Materials Integrity Section.

A key development in 1987 has been the identification of a technique for the nondestructive measurement of the hydrogen content in pressure tube materials. Further work to develop this technique for use on power reactors will take place in 1988. Another milestone of 1987 was that the highest fluence specimens in the Dido irradiation growth program have exceeded the average fluence levels in central channels of Pickering 3.

For 1988, The Materials Integrity Section will increase its' capabilities and expertise through the transmission electron microscopy of irradiated materials. These studies will increase the knowledge of how the microstructure of pressure tube materials are changed by exposure to different fast neutron flux spectra. Specimens from the Dido program and from surveillance

pressure tubes will be examined. Continued studies in blisters, delayed hydride cracking and irradiation deformation will help support the OH CANDU program by the development of an understanding of how metallurgical parameters and operating variables can influence each process. The ultimate goal is to produce predictive capabilities that can determine the integrity of pressure tubes as a function of operating time and conditions. Also, work in support of the development of improved components (calandria and pressure tubes) will continue with the main emphasis on TG3 materials.

In support of thermal plant, the Materials Integrity Section can only consider involvement with the EPRI funded corrosion-fatigue study of boiler tube materials. There are no resources currently available to expand our involvement in areas of needed research into thermal plant materials.

To ensure that Ontario Hydro maintains its own expertise in the metallurgy of nuclear and thermal plant components, it is important that the Materials Integrity Section is provided the necessary resources to perform the planned work. Also of importance is the need for easy access to specific facilities in order to develop expertise to meet future demands. In particular, advanced electron optics are required to provide an understanding of how metallurgical structure changes with exposure to service conditions to establish end-of-life properties and to determine if the metallurgical structure could be changed to extend the service life of key components. Also for nuclear plant studies, the staff requires easy access to hot cells to perform tests on irradiated materials. The hot cell capability envisaged is not an extensive complex, but more of a shielded cabinet type assembly for handing material that has been out of reactor for several years.

2.0 MATERIALS INTEGRITY SECTION PERSONNEL

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3.0 DEFORMATION MECHANISMS UNIT - 1987 REVIEW

3.1 Reactor Data Analysis

3.1.1 Pressure Tube Sag Deflection

(N. Badie and R.A. Holt)

A comparison of the sag predictions of the two pressure tube design equations provided by AECL, CRNL-1993 (issued in 1979) and CRNL-4000 (issued in 1985) was carried out using the Ontario Hydro computer code CDEPTH. Until now CRNL-1993 has been used for all design calculations, however, no systematic assessment has been made of the accuracy of sag predictions relative to power reactor data. CRNL-4000 fits a wider range of experimental data and hence is expected to provide better predictive capability.

When the same input conditions are used CRNL-4000 consistently predicted greater sag deflection than CRNL-1993 as shown for P3 J09 in Figure 1. The location of the maximum sag was similar for both equations. When the predictions were compared to in-reactor measurements of channels in the CNS periodic inspection program in Pickering Unit 3 and Bruce Unit 2, for two fluence levels, the following observations were made:

- Both CRNL-4000 and CRNL-1993 overpredicted the maximum sag of Pickering 3 channels at both fluences. Also, the differential sag was overpredicted by both equations.

- Both CRNL-4000 and CRNL-1993, on average, overpredicted the maximum sag of Bruce 2 channels at both fluences. Also, CRNL-1993 underpredicted the differential sag while CRNL-4000 was in close agreement with in-reactor measurements in this respect.

- The sag deflection predictions were sensitive to the calandria tube design equation. Since an uncertainty of the order of a factor of two exists in the derivation of the calandria tube creep constant, this parameter could be adjusted to tune the predictions to the in-reactor measurements, independently of the pressure tube design equation, Figure 1. Hence, sag deflection calculations are not a good test of the pressure tube design equations.

- The sag predictions were also seen to be sensitive to reactor dimensional and operating data and to the measuring conditions. This data is currently being reviewed to ensure that CDEPTH inputs reflect reactor conditions as accurately as possible.

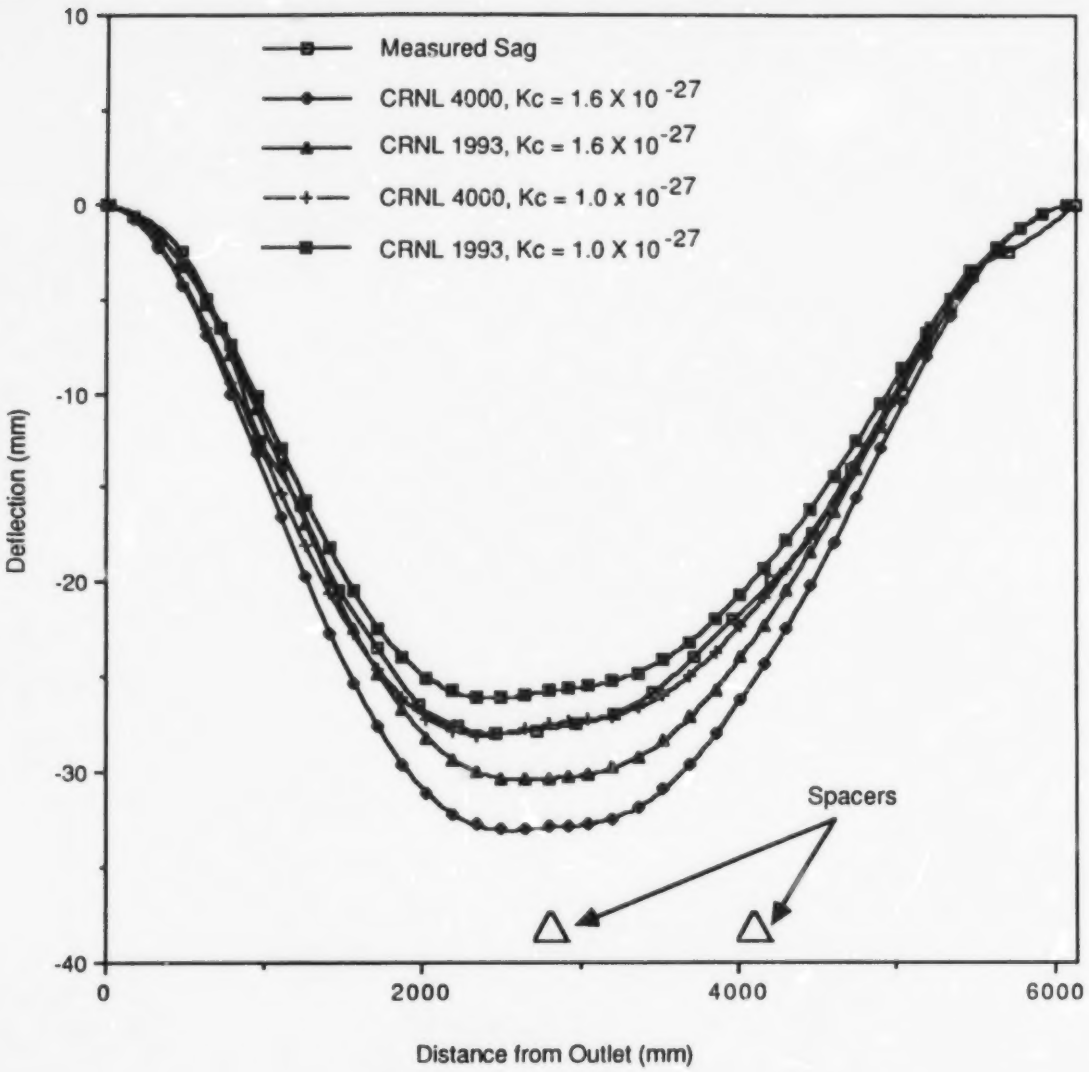


Figure 1 Calculated sag deflections and measurements for P3 J09 after 84,000 EFPH of operation. Kc is the value of the calandria tube creep constant assumed in $m^2/n/MPa/h$, E>1 MeV.

3.1.2 Pressure Tube/Calandria Tube Contact
(N. Badie and R.A. Holt)

Because of the high deuterium concentrations observed near the outlet end of P3 L09, removed for post-irradiation examination in August 1987, the development of pressure tube to calandria tube contact in channels with misplaced spacers has become a major concern. Hence, a new study was undertaken to assess contact development. The contact predictions of the computer code CDEPTH were compared with contact length measurements made during post-irradiation examination of four channels removed from Pickering A and Bruce A for surveillance (P3 L09, P3 J09, P4 K10 and B2 P12) plus the ultrasonic indications of contact obtained by CNS during in-service inspection of P3 M12.

The calculations were done using both available pressure tube design equations (CRNL-1993 and CRNL-4000). In all cases under study, CRNL-4000 predicted earlier initial contact and more rapid spreading of the contact compared to CRNL-1993. Also, CRNL-4000 predicted the contact area to spread more rapidly. The predicted time of initial contact and subsequent spread of contact were found to be rather insensitive to the calandria tube design equations. Figure 2 illustrates these points for the predicted spreading of contact for channel P3 J09.

When the predictions for the data set were compared with measurements, CRNL-4000 was found to give better agreement, on a simple average, with reported lengths and locations of contact. This is illustrated in Table 1.

TABLE 1

COMPARISON OF MEASURED CONTACT LENGTHS AND POSITIONS WITH THOSE CALCULATED USING CDEPTH

Channel	Center of Contact (m from core centerline)			Length of Contact (m from core centerline)		
	measured	calculated CRNL-1993	calculated CRNL-4000	measured	calculated CRNL-1993	calculated CRNL-4000
P3 J09	1.48	1.37	1.43	0.44	0.36	0.59
P3 L09	0.87	0.85	0.92	1.19	1.11	1.52
P3 M12	0.45	0.15	0.23	2.95	2.58	2.96
P4 K10	-0.34	-0.28	-0.30	0.19	0.29	0.34
B2 P12	-0.89	-0.60	-0.63	1.29	0.78	0.92

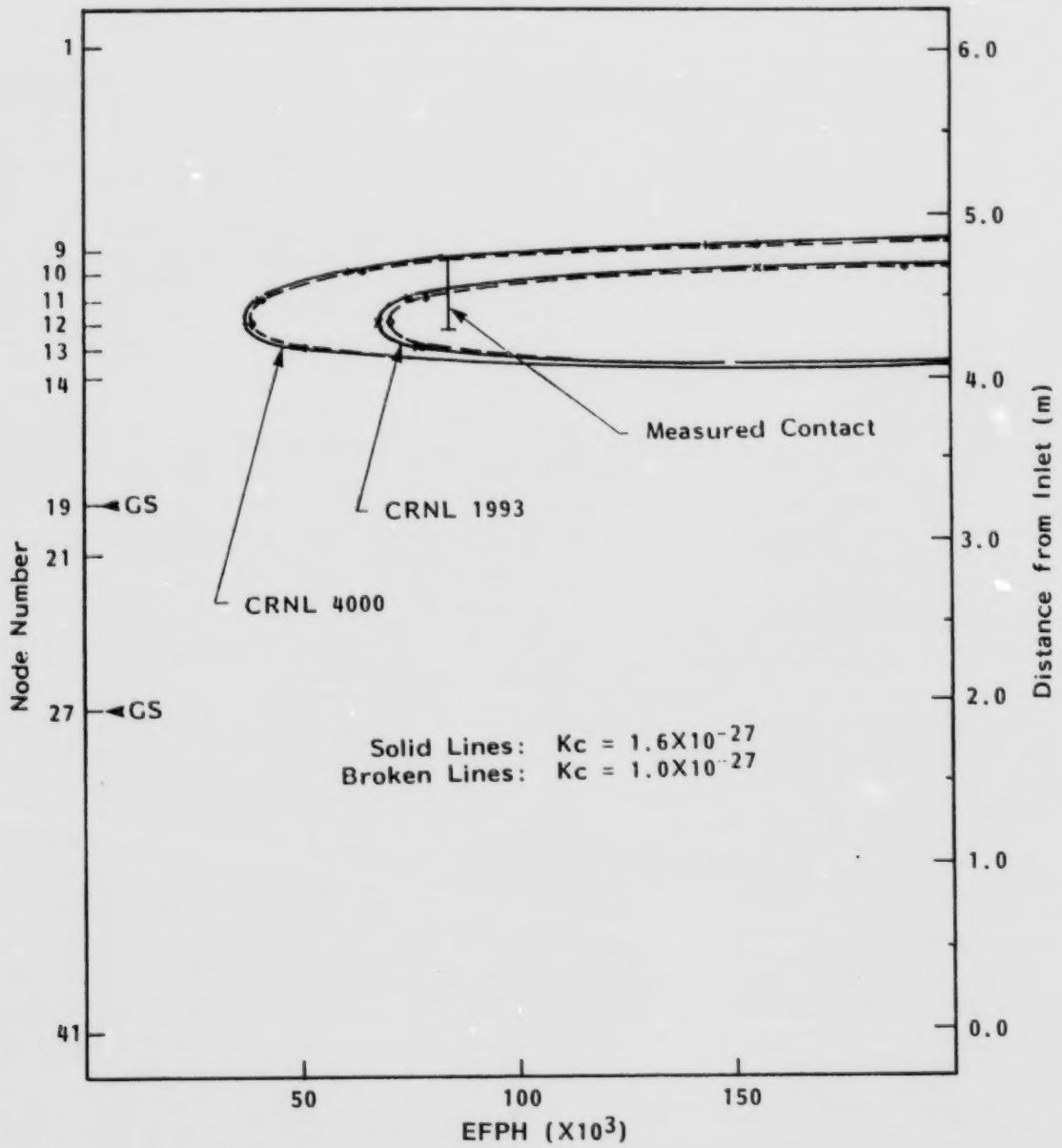


Figure 2 Calculated development of contact with operating time for P3 J09 and measured extent and location of contact after 84,000 EFPH. Kc is the value of the calandria tube creep constant assumed in $m^2/n/MPa/h$, E>1 MeV.

Because the contact time and spreading are insensitive to the calandria tube creep rate assumed, comparison of contact predictions with measurements, together with other data from diametral and elongation measurements and predictions, can provide a test of the accuracy of the pressure tube design equations. Once the pressure tube design equation is selected on this basis, and after review of the operating and measuring conditions, the calandria tube design equations could be tuned to the sag deflection data for each reactor.

3.1.3 Modelling Contact Spreading

(R.A. Holt and N. Badie)

A program was initiated to provide an algorithm to calculate the spread of pressure tube/calandria tube contact for the purposes of probabilistic assessment of blister formation, the use of CDEPTH being too unwieldy for this purpose. A functional relationship for contact spreading was derived from CDEPTH calculations. Parameters were then selected to give a fit to the five sets of contact length data referred to above with initial contact locations and times based on the existing probabilistic model.

The spread of contact is taken to consist of two components, an initial rapid stage, during which a considerable amount of elastic spreading occurs, followed by a slow spread of contact governed by creep.

In the model, the initial, rapid, spread of contact is taken directly from CDEPTH calculations of the extent of contact after 10,000 EFPH. This is given by:

$$S = S_o(t/10)^{a \cdot \ell_o}$$

where: S is the half length of contact (ie, the distance between the furthest extent of contact and the next support),
S_o is the half length of contact after 10,000 EFPH,
t is the time since contact in 1000's of EFPH,
ℓ_o is the initial unsupported half length (ie, the distance from the point of initial contact to the next support),
a is a constant,

and: $S_o = b\ell_o^q$

where: b and q are constants whose values depend on whether the next support is a spacer or the end fitting.

The later creep controlled spread of contact is assumed to occur at a rate:

$$dS/dt = -KS^n$$

and hence:

$$S = \ell_o - (K't + \ell_o^m)^{1/m}$$

where: n is a constant,
m = 1 - n,
K is a constant,
K' = -m.K,
ℓ_o' = ℓ_o - S_o.

According to the CDEPTH calculations, n has a value of about 5, and this depends on the location relative to the inlet, on whether the contact is spreading towards a spacer or an end fitting, on the orientation of the pressure tube, on the flux and temperature profiles etc. However, in comparisons with measured contacts for the five channels cited above, constant values of n and K give a good fit to the data, Figure 3. The calculated spread of contact is shown for 3 channels in Figure 4. This model has been used in the most recent assessments of blister probabilities in Pickering 3 and 4.

3.1.4 Pressure Tube Elongation
(R.A. Holt)

The pressure tube elongation data base compiled by NSD was obtained by Research Division and used to assess the contribution to the variability in elongation of metallurgical factors. The variation in the elongation rates of tubes made by the standard manufacturing process in the central region of Bruce 2 have been analyzed. The variation in elongation for a given set of measurements is assumed to consist of five components:

- that due to systematic variations in the measurement down the reactor face.
- that due to tube-to-tube flux variations
- the "intercept" variation, which includes random measurement errors and variations in installed length of the tubes
- that due to temperature variations
- "true" tube-to-tube variations in elongation rate due to metallurgical variations amongst the tubes.

The random measurement errors, temperature and "true" variations account also for the variation in the elongation rate of the tubes, relative to fast neutron fluence. The analysis shows that the true tube-to-tube variation in elongation rates is 11 - 14 %, compared with an observed variation of 17%. This would lead to an apparent variation in elongation of 21-24 mm in 30 years of operation. Based on this analysis, offcut were selected and obtained from CNS for metallurgical evaluation to assess the cause of the variations.

3.2 In-Reactor Deformation Mechanisms

3.2.1 Dido Irradiation Experiment
(R.A. Holt)

The irradiation program being conducted in the Dido reactor at Harwell to study the microstructure and temperature dependence of pressure tube growth continued throughout 1987. The fluence of the most highly irradiated specimens (from a Bruce A pressure tube) exceeded the axially averaged fluence in a central channel of P3 in mid 1987.

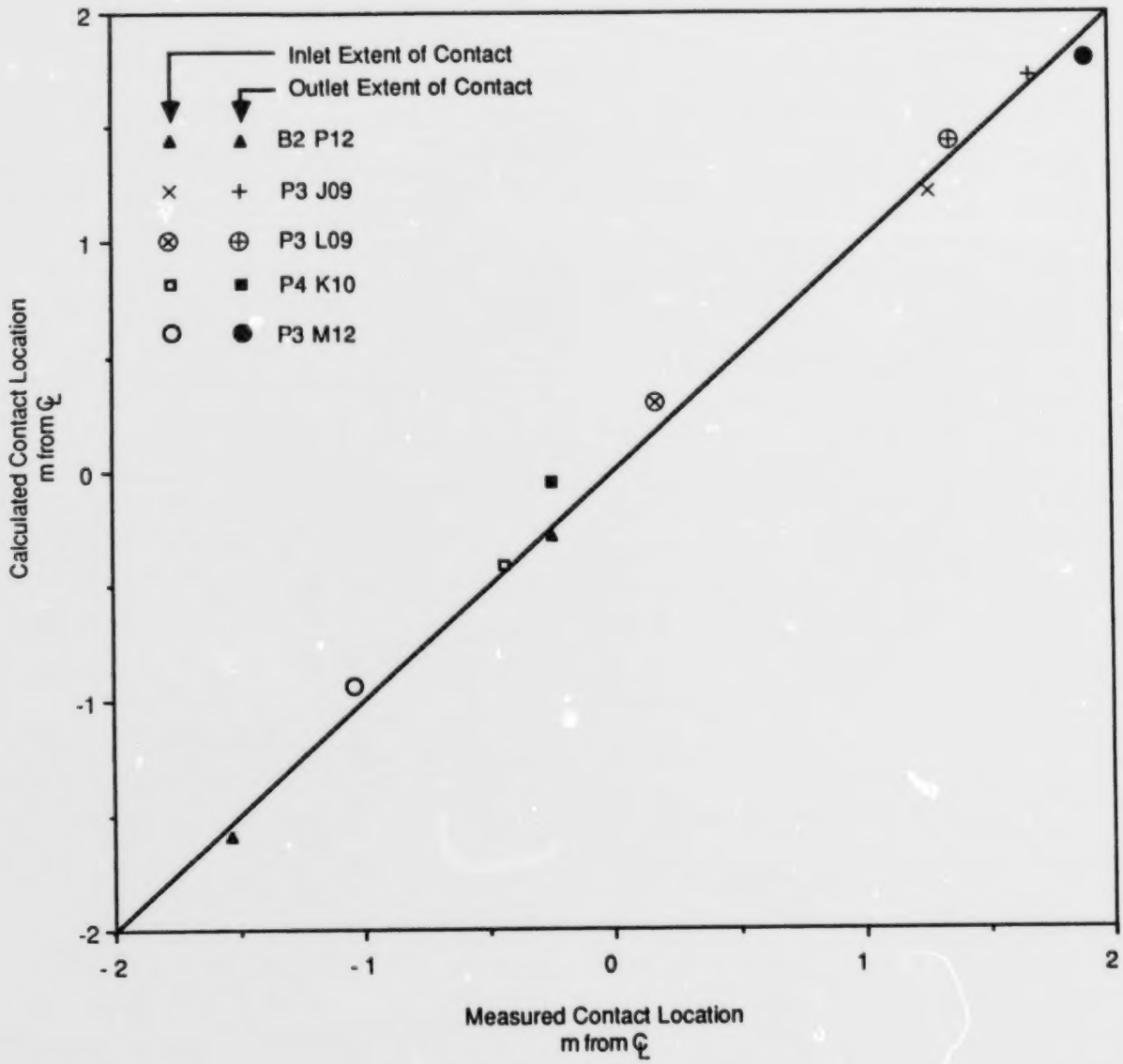


Figure 3 Comparison of calculated contact locations (extent of contact towards channel inlet and outlet) using the contact spreading model vs measured extent of contact.

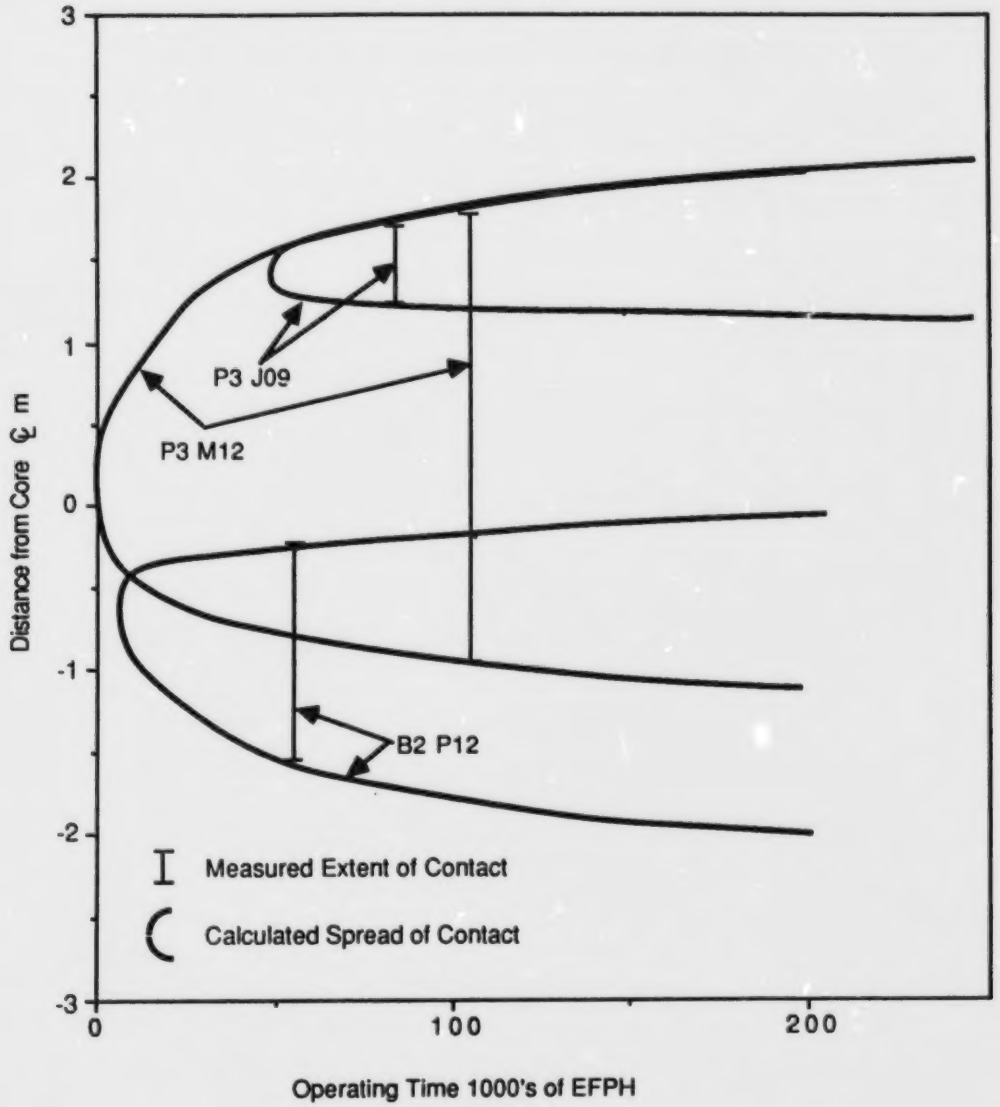


Figure 4 Calculated spread of contact vs operating time for three fuel channels compared with extent of contact measured during in-service inspection (P3 M12) or post-irradiation examination (P3 J09 and B2 P12).

3.2.1.1 **Rate of Irradiation Growth**

The rate of irradiation growth observed in Bruce A pressure tube material in Dido is 1/4 of the rate of elongation of Pickering A and Bruce A pressure tubes and about 1/3 of the rate of irradiation growth rate given by current design equations. Several approaches have been initiated to resolve this discrepancy.

Growth specimens machined from two channels removed for surveillance (P4 K10 and B2 P12) were prepared at CRNL and shipped to Harwell for inclusion in the program. Both sets of specimens have fluences of 6-7 $x10^{25}$ n/m^2, E>1 MeV. Irradiation of the specimens commenced November 1987. Any discrepancy between the growth rate of these specimens, and the rate of specimens irradiated from zero fluence without stress in Dido would be evidence of interaction between creep (stress driven deformation) and growth, and would necessitate a change in the form of the design equations.

Measurements of the fast neutron flux in both the surveillance tube specimens and growth specimens is being undertaken by the radiation dosimetry group at Harwell. The $^{93}Nb^{m}$ concentration is being measured to produce an estimate of the fast flux >0.5 MeV. The results will provide a normalization of fast neutron flux in Dido with that in a CANDU pressure tube.

Transmission electron microscopy was carried out on specimens of P4 K10 and B2 P12, taken from a section adjacent to the growth specimens. Preliminary analysis shows that there is a fairly high residual density of dislocations with <c+a> Burgers vectors and the presence of Nb rich precipitates in the alpha phase. These observations will be compared with observations on growth specimens removed from Dido. Thin foil blanks were prepared from three of the growth specimens late in 1987 and will be shipped to OHRD for final preparation and examination.

3.2.1.2 **Temperature Dependence of Growth**

The effect of a downward change in temperature on the growth in the longitudinal direction of standard and modified (Task Group 3 Route 1) pressure tube material has been investigated. The irradiation temperatures of specimens previously irradiated at 583 and 623 K were reduced to 553 K. After the temperature change, a positive transient strain is observed, the magnitude of which increases with the magnitude of the temperature change as shown in Figure 5. This is approximately the reverse of the behaviour previously reported for an increase in irradiation temperature. When the prior irradiation temperature was 623 K, the subsequent growth curve appears to saturate at a strain lower than that which would have been achieved had the specimen been irradiated from zero fluence at the lower temperature. In a specimen initially irradiated at 553 K to $1.7x10^{25}$ n/m^2, E>1 MeV, then irradiated for $1.8x10^{25}$ n/m^2, E>1 MeV at 623 K before being returned to 553 K, saturation appears to occur at a lower strain relative to continuous irradiation at the lower temperature.

The transient strains resulting from a temperature increase from 583 K (reported previously) have been analyzed further. With reference to a baseline of continued irradiation at 583 K the TG3 material and the standard pressure tube material demonstrate remarkable similarity. The sign, magnitude and duration of the differential strains are similar despite the fact that the anisotropy of the baseline growth of the two materials at 583 K is very different. In both cases the magnitude of the differential strain increases with fluence Figure 6. Apparently the source of

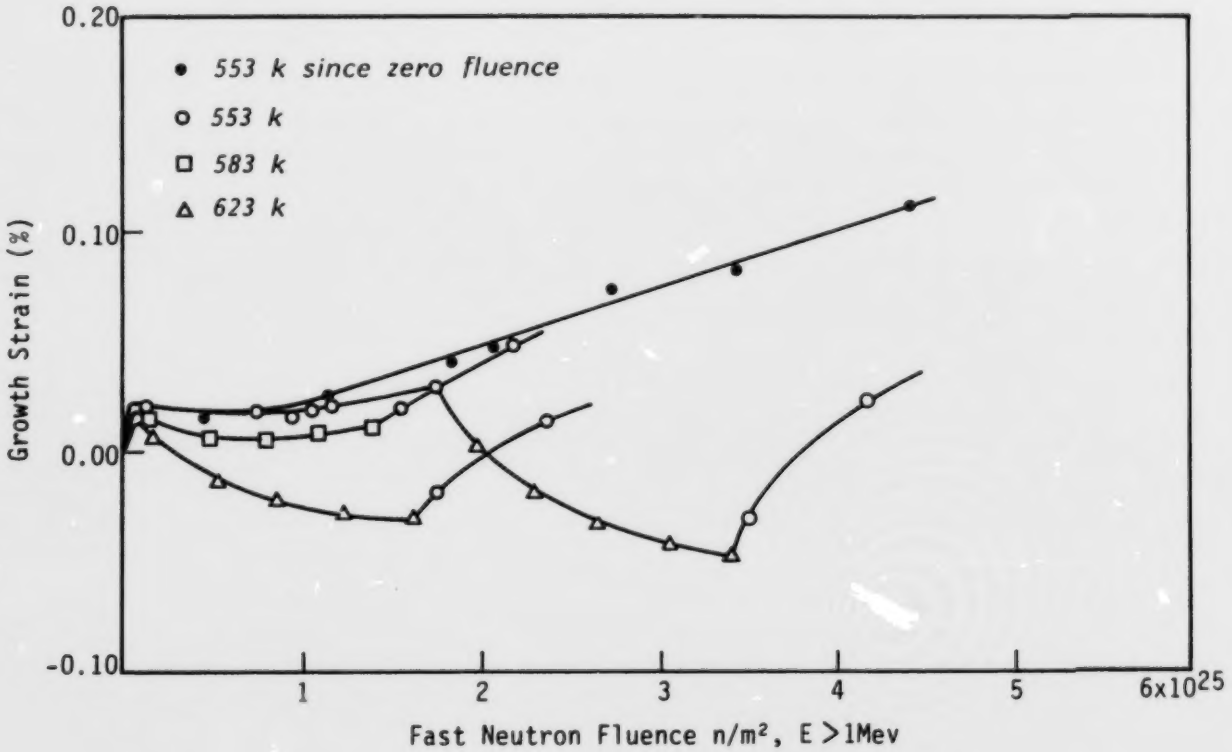


Figure 5 Effects of increases and decreases in irradiation temperature on the growth strain in the longitudinal direction of standard pressure tube material irradiated in Dido at a fast neutron flux of 6-7x10^{17} n/m^2/s, E>1 MeV compared with continued irradiation at 553 K.

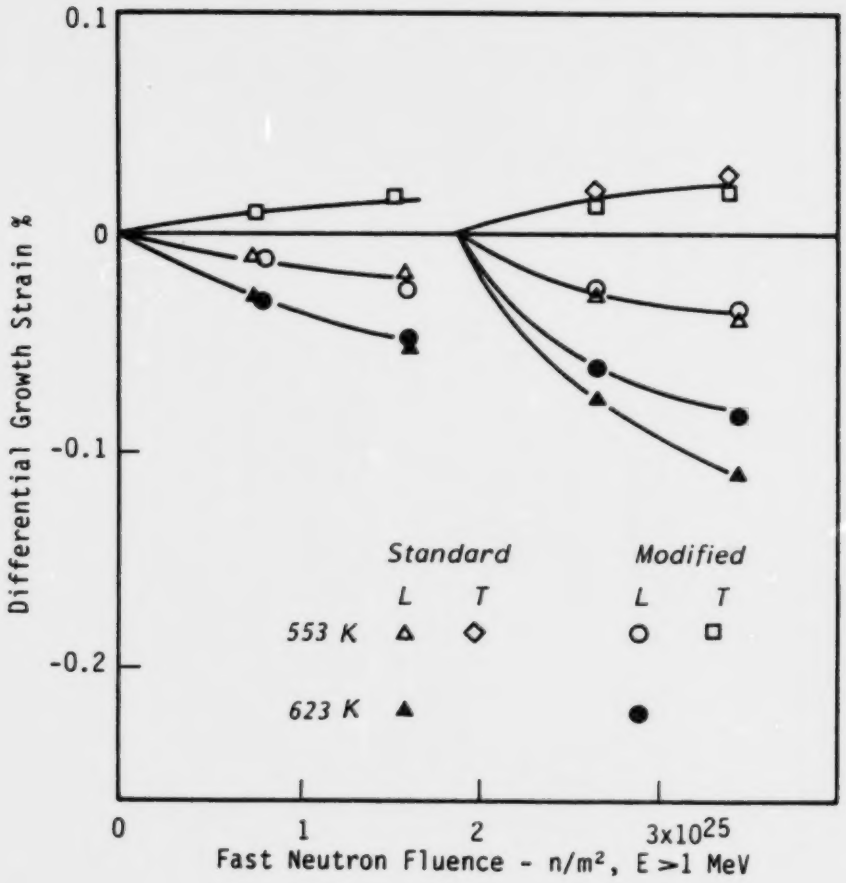


Figure 6 Incremental growth strains compared with continued irradiation at 553 K for increased irradiation temperatures from zero fluence, and after about 2×10^{25} n/m^2 E>1 MeV (L indicates longitudinal direction and T transverse direction).

the transient is the same in both materials. The anisotropy is approximately the reverse of that observed in the initial stages of growth, and in the steady-state growth of the standard material, but different from that of the modified material. This suggests that the processes involved occur predominantly within the alpha-grains, and do not involve the boundaries to a large extent.

These results indicate that the effects of power maneuvering on deformation may be complex and depend on the detailed temperature histories but not on the fabrication route of the pressure tube.

3.2.2 Mechanisms of Irradiation Growth
(R.A. Holt)

A mechanistic model for irradiation growth of zirconium alloys is being developed to help understand the results of the irradiation growth program in Dido and assess their implications for power reactor pressure tubes. The model includes recent theoretical concepts and the known features of the microstructure. In particular, it allows for the contribution to irradiation growth from anisotropic point defect migration in addition to the contribution from the usual "size interaction" between self-interstitial atoms and dislocations, and includes, as potential point defect sinks, <c+a> and <a> network dislocations introduced by working prior to irradiation, <a> type loops on prism planes, <c> component loops on basal planes and high angle grain boundaries.

The model is normalized to irradiation growth data obtained for annealed and cold-worked Zircaloy-2 at 353 and 553 K by workers at the UKAEA and is tested against results obtained in ATR by AECL and by OHRD in the Dido growth program.

The model is able to account for the effects of dislocation density on growth at 353 K and the acceleration in growth of annealed Zircaloy-2 at 553 K when <c> component loops appear at high fluences ("growth breakaway"). When the model is normalized to the growth rate of cold-worked Zircaloy-2 at 553 K and normal vacancy migration is assumed, the "anomalous" growth behaviour of the Task Group 3 Route 1 (TG3 R1) pressure tube material - ie, negative growth in both the axial and transverse directions - is predicted. The behaviour is predicted to be a function of both the original network dislocation density and the grain thickness as shown in Figure 7. This could explain why standard pressure tube material (0.3-0.4 μm thick grains, 5-7x10^{14} m^{-2} dislocation density) exhibits "normal" growth while the TG3 R1 material (0.2-0.3 μm thick grains, 2-3x10^{14} m^{-2} dislocation density) exhibits "anomalous" behaviour.

An important implication of this result is that the "anomalous" behaviour - which is beneficial for reactor operation - would be true "steady-state" growth, and not a transient associated with relaxation of intergranular constraints as previously supposed.

The model is currently unable to account, in a self consistent way, for the low temperature sensitivity of irradiation growth between 353 and 553 K possibly due to the lack of understanding of the point defect properties of alpha zirconium.

3.2.3 End-of-Life Deformation Characteristics
(R.A. Holt)

Current projections of the deformation of pressure tubes in CANDU reactors at the end of life are based on extrapolation of the deformation rates observed in the first 10 years of operation. There are no good theoretical grounds to assume that such an extrapolation is valid. In fact, based on our current understanding of irradiation induced deformation processes it is likely that

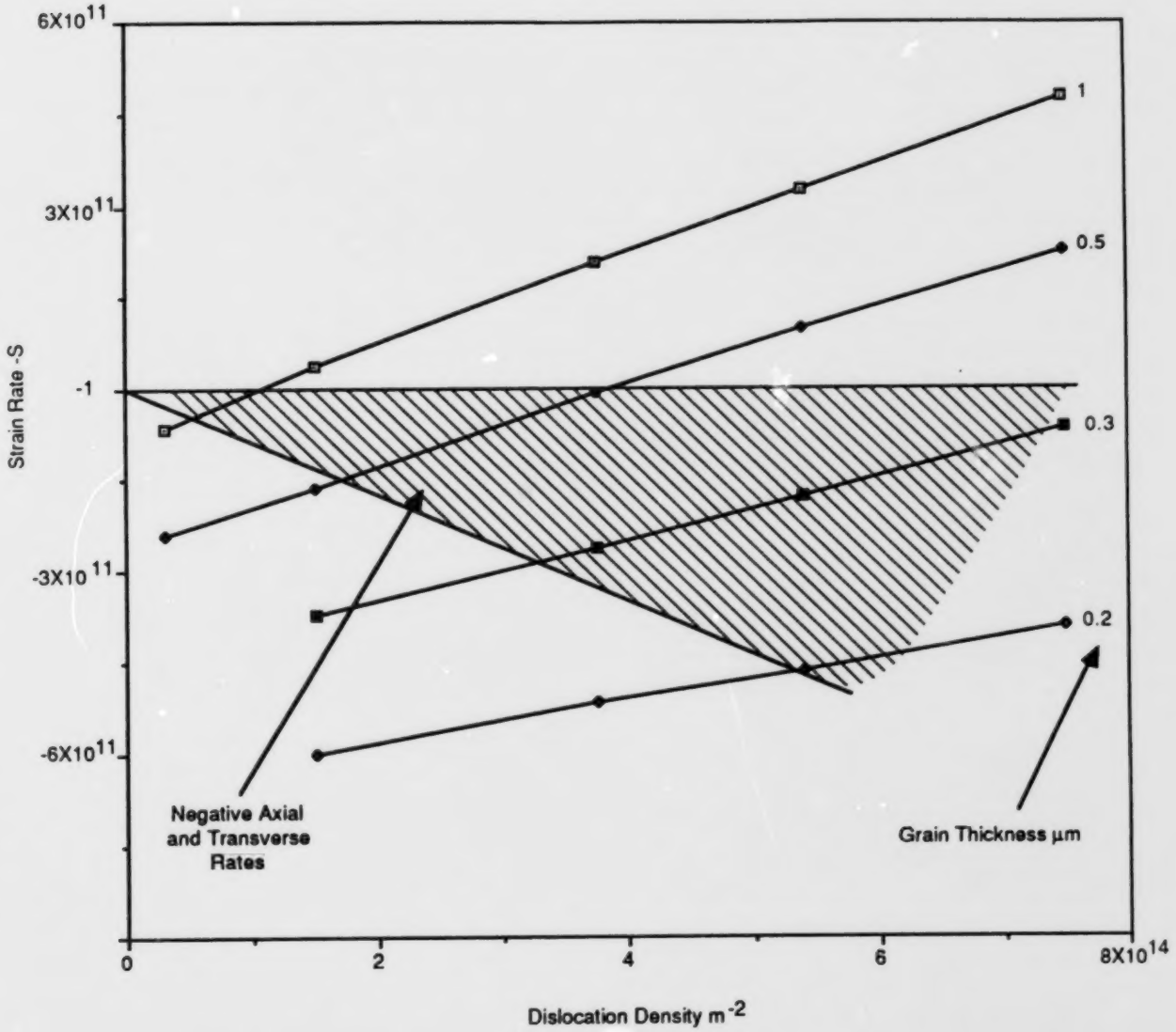


Figure 7 Calculated rates of longitudinal irradiation growth as a function of grain thickness and dislocation density at a displacement rate of 5×10^{-8} dpa/s showing the possible existence of a regime in which both axial and transverse growth rates are negative. This is found experimentally for TG3 R1 pressure tube material.

acceleration or deceleration of the deformation rates will occur. This could be gradual, in response to gradual evolution of the dislocation substructure with fast neutron fluence, or sudden, as in the case of accelerated irradiation growth of annealed Zircaloy at 570 K.

Since long-term decisions on fuel channel maintenance are made on the basis of the extrapolated rates, experiments are required to assess the validity of the extrapolations. Irradiation induced deformation rates are proportional to the fast neutron flux, hence, accelerated experiments can be performed in high flux test reactors to evaluate these possibilities.

Two experimental reactors in France (Osiris at Saclay, and Siloe at Grenoble) have very high fast neutron fluxes (i.e $2x10^{18}$ n/m^2/s, E>1 MeV compared with $6x10^{17}$ in Dido, at Harwell and in the newest facility in NRU at Chalk River and $2\text{-}3x10^{17}$ in CANDU reactors).

The French facilities offer the possibility of performing experiments to evaluate the effect of very high ("end-of-life") fast neutron fluences on pressure tube properties in a reasonable time scale, at a reasonable cost. For example, the costs, and schedules for performing irradiation growth experiments to end-of-life fluences are shown in Table 2 in comparison with similar estimates for extending the current growth experiments in Dido to very high fluences.

TABLE 2

COMPARISON OF IRRADIATION COSTS IN DIDO AND THE FRENCH HIGH FLUX FACILITIES SILOE AND OSIRIS

	DIDO	SILOE/OSIRIS
Irradiation to 1.5 x 10^{26} n/m^2		
Neutron Cost	k$ 2,507	k$ 666
Hardware	k$ 80	k$ 203
Measurements [3]	k$ 170	k$ 53
TOTAL COST	k$ 2,757	k$ 922
Completion Date	1994	1992
Irradiation to 3.0 x 10^{26} 26 n/m^2		
Neutron Cost	k$ 7,207	k$ 1,332
Hardware	k$ 240	k$ 406
Measurements [3]	k$ 454	k$ 106
TOTAL COST	k$ 7,901	k$ 1,844
Completion Date	2008	1996

[1] 40 growth specimens
[2] 36 growth and 6 creep specimens
[3] Assumes measurements made every 3 x 10^{25} n/m^2

The neutron energy spectra in Dido and both French reactors are reasonably well characterized, although there is some question as to how the spectra compare with that experienced by a pressure tube in a CANDU reactor in the energy range 0.1-1.0 MeV (about 40% of the displacement damage is believed to be caused by neutrons with energies in this range in a CANDU pressure tube). The C.E.A. have a great deal of expertise in the area of in-reactor experimental inserts, and knowledge of the conditions experienced by the specimens would be adequate for the types of experimental conditions envisaged.

Funding has been approved to perform accelerated creep and growth tests on standard and modified pressure tube materials in one of the French reactors. Discussions have been held with the CEA, a preliminary proposal has been received from the CEA and the technical details of the experiments are under negotiation. The preliminary schedule calls for the irradiation to start late in 1988.

3.2.4 Thermal Creep

(R.A. Holt, M. McGraw and Z. Grozdanovski)

According to the most recent equation for in-reactor deformation of pressure tubes (CRNL-4000), in-reactor "thermal" creep is a significant component and this component which accounts for the pronounced peak in diametral strain near the outlet ends of pressure tubes with outlets at their west ends Pickering Unit 3. This component contributes significantly to the sag deformation of the pressure tubes, but its anisotropy and stress dependence have not been studied. A program to investigate the anisotropy, stress dependence and dependence of creep rate on position along the pressure tube has been undertaken. Initial results suggest large anisotropy of the thermal creep rate at 570 K at stresses above 300 MPa.

3.3 Pressure Tube Development

3.3.1 Assessment of Production TG3 Pressure Tubes

(R.A. Holt)

The first production batch of pressure tubes made to the Task Group 3 Route 1 (TG3 R1) specification demonstrated marginal tensile strength. This is a marked contrast to the pilot order of TG3 R1 tubes which were very strong compared with standard tubes. Examination of the microstructures of the tubes by transmission electron microscopy indicated that a small amount of recrystallization had occurred during stress relief, suggesting that the cold worked microstructure of the production tubes is somewhat less stable than that of the pilot order of tubes. Several other factors were identified which may have contributed to the lower strength in the production tubes, ie;

- the specified level of allowable oxygen concentrations was reduced by 300 ppm compared with the pilot order
- the beta-quench used to refine the grain structure of the extrusion billet was performed on the solid forgings rather than on the hollow billets
- the billets were preheated for a period of 7-8 hours prior to extrusion compared with three 1/2 hours for the pilot order
- the effective stress relief time was longer because of the slow heat-up and cool-down times used in production

An experimental program was undertaken to identify the functional relationship between tensile strength and stress relief conditions to allow selection of an appropriate modified stress relief treatment for tubes in process.

3.3.2 Elevated Temperature Properties for Zr-2.5 Wt% Nb Pressure Tubes
(E.T.C. Ho and M. McGraw)

A test program was initiated to examine, in detail, the tensile properties of Zr-2.5 wt% Nb PT's made to the Task Group Three (TG3) specifications. Materials in the as cold-worked (CW) condition from several TG3 tube offcuts were obtained from Nu Tech. Axial tensile specimens were machined using the above offcuts. Results obtained to date indicate that the 0.2% offset yield and tensile strengths of the weaker front-end offcuts in the as CW condition are about 497 and 555 MPa, respectively. These values are well above the yield and tensile strength specifications for the PT's which are 331 and 479 MPa, respectively. However, a stress relief treatment of six hours at 500°C reduces the yield and tensile strengths of the CW material significantly. More importantly, the tensile strength of the stress-relieved front-end offcut materials was slightly below specification.

The remaining material will be used to assess the strength reduction of the as CW material as a function of heat treatment temperature and time at temperature.

3.3.3 End-of-Life Fracture Properties of TG3 Pressure Tubes
(R.A. Holt)

Nine Task Group 3 pressure tubes are installed in Bruce Unit 8, and TG3 R1 pressure tubes are currently in production for retubing and replacement purposes. TG3 tubes were qualified for reactor service, based on their short term properties (up to two years equivalent reactor exposure). Hence, the data base on fracture properties is much less extensive than that for standard pressure tubes which have been removed from service with up to 15 years of exposure.

To assess the long-term fracture properties of the TG3 R1 material, a program has been planned to irradiate fracture toughness, delayed hydrogen cracking and tensile specimens to end-of-life fast neutron fluences in one of the French high flux facilities described in Section 3.2.3. A preliminary proposal has been prepared by the CEA and technical details are being negotiated.

3.3.4 Phase Transformation (β(b.c.c.) $\rightarrow$ α(h.c.p.)) in Zr-2.5 Wt% Nb Alloys
(V. Perovic and R. Jarochowicz)

It has been demonstrated that the α/β interface of the β quenched or slowly cooled alloy from the β_{Zr}-phase field contain dislocation arrays. One of the arrays is composed of remarkably straight and regularly spaced dislocations with $[111]\|\|[\bar{2}113]$ Burgers vector. The other, less regular array of dislocations was analyzed to have $[\bar{1}11]\|\|[\bar{1}2\bar{1}0]$ Burgers vector. The observation of very regular arrays of dislocations of the α/β interface suggests that they are related to the phase transformation. It is found that the most illuminating way to discuss the origin of these dislocations is in terms of the concept of an invariant line. This line is defined as a line which is unaffected by the phase transformation. The vectors parallel to this line are unstretched and unrotated by the transformation and accordingly there is no misfit between the two lattices along this line. Thus, if the precipitate/matrix interface contain this invariant line direction, one condition for a low-energy interface is met. It is well known that whenever the three principal

strains describing the structural part of a phase transformation include two strains of opposite sign, there will be a set of vectors which remain unextended (although rotated) during transformation. If one of the unextended vectors is rotated back to its initial position by a rigid body rotation, then this vector is on the invariant line of the transformation. The best matching between the observed dislocation line direction and calculated invariant line is obtained when two consecutive rotations are performed. The first rotation (5.26°) around [110]||[0001] direction rotate two lattices from the Pitsch-Schrader to Burgers relationship and the small second rotation (~1°) around $[\bar{1}11]\|[\bar{1}2\bar{1}0]$ axis is required to complete the transformation. The double rotation is also consistent with the <a+c> Burgers vector of the interface dislocations. The second less regular dislocations along the α/β interface are analyzed as <a> type dislocations. The dislocation substructure of the α/β interface, established during β_{Zr}-treatment and further developed during extrusion, is retained in the pressure tubes after cold working and stress relieving. Thus, it is clearly demonstrated that one of the characteristic substructure features of Zr-2.5 wt% Nb pressure tubes is established in the early stages of the pressure tube fabrication, ie, β_{Zr}-treatment and extrusion. Most of the experimental and analytical work is nearly completed and the main effort at present is focused on the summary of these results.

3.3.5 Dislocation Substructure

(V. Perovic and R. Jarochowicz)

The nature of the dislocation substructure formed at the α/α and α/β boundaries in Zr-2.5 wt% Nb pressure tube alloys has been studied by electron microscopy. This study is a part of an ongoing program and some results are already published internally and externally. It is shown that the character of the boundaries is established by the extrusion operation, and subsequent cold working and/or annealing treatments modify only the misorientation and nature of the boundaries but not the scale of the substructure. The dominant boundary structure introduced by extrusion contains arrays of <a>, <a+c> and <c> dislocations. A number of different arrays of dislocations were observed and more complete analysis of these boundaries and the electron diffraction contrast of the component dislocation is currently in progress. The experimental, part of this program is expected to be completed in the near future, and the main effort at present is focused on the summary of these results.

3.3.6 Evaluation of Pilgered Tubes

(G. Shek and H. Seahra)

Current Zr-2.5 wt% Nb pressure tubes are cold worked by plug drawing. Pilgering is being evaluated as an alternative method of cold working the tubes. Tube samples from three pilgered tubes were obtained from AECL-CO for testing at OHRD as part of the overall evaluation program. The tests required to be done at OHRD were longitudinal tensile tests at room temperature and 300°C; texture and dislocation density measurements; grain size and phase characterization; and threshold orientation stress for radial hydrides.

The test results show that the pilgered tubes are similar to the current cold drawn tubes in terms of strength, texture, dislocation density, microstructure and threshold stress to form radial hydrides. There is also less difference in tensile strength and dislocation density between the front and back end in the pilgered tubes than in current cold drawn tubes.

However, the delayed hydride cracking tests done by AECL-RC showed some inconsistency in K_{1H} values in samples taken from the pilgered tubes. Two out of 10 tests had low K_{1H} values and there was no obvious reason that could account for the two low K_{1H} values. Based on the DHC test results, the pilgered tube was rejected as a replacement for a surveillance tube removed from Pickering 3.

A program was proposed by AECL to determine whether variation in residual stresses is the reason for the low K_{1H}.

3.4 Materials Properties of Zircaloy-4 Fuel Cladding
(E.T.C. Ho and A. Donner)

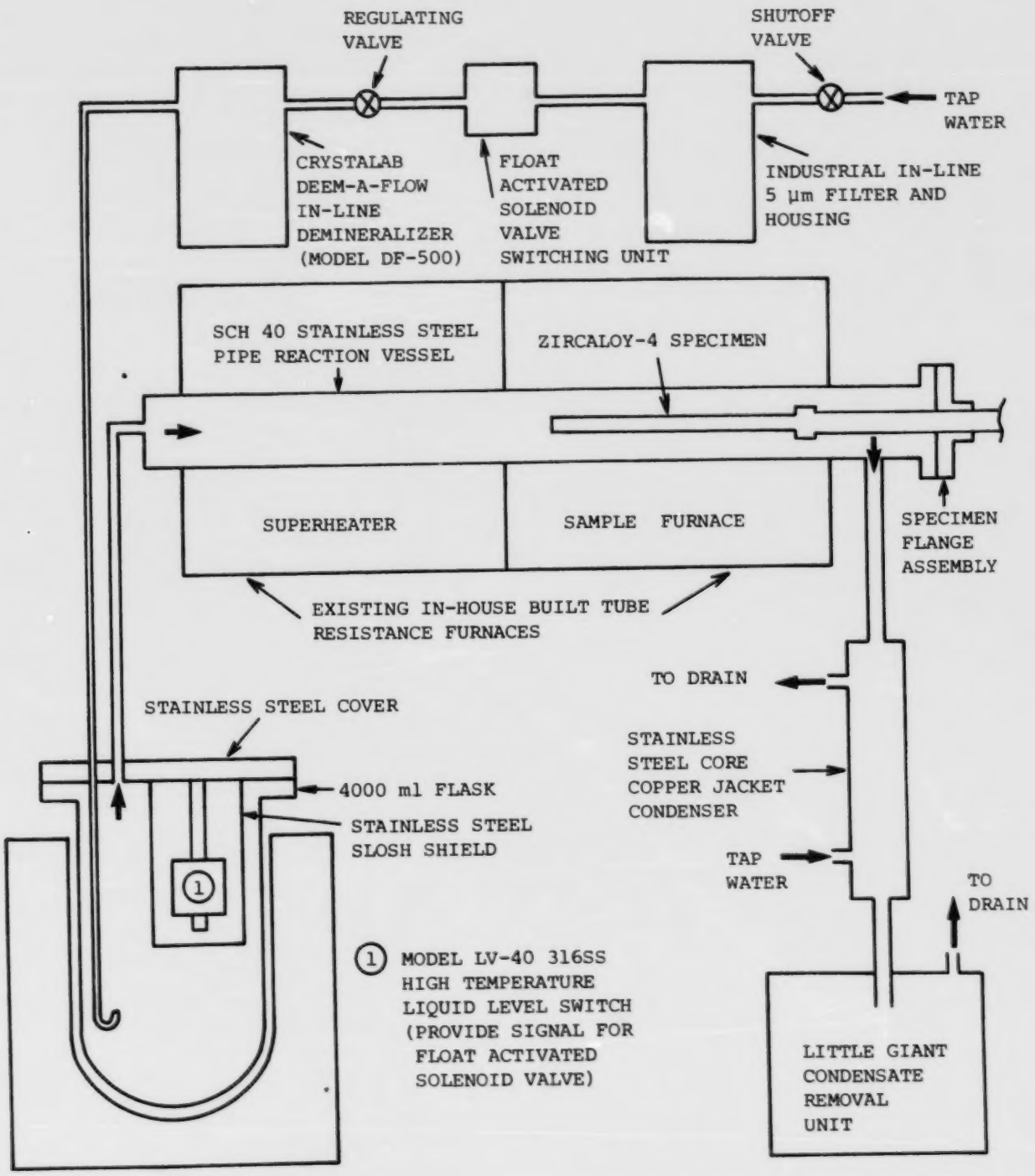
Recent results obtained from a test program carried out at OHRD show that braze heat affected zone (HAZ) cladding material oxidizes 1.2-2.0 times more rapidly than parent material in an air environment over the temperature range 600-850°C. More importantly, comparison of the test results with available steam oxidation data, such as those reported by Leistikow et al, indicate that the oxidation of parent Zr-4 may be more aggressive in air than in steam. Since the analysis of a number of post-shutdown accident scenarios and fuel handling incidents requires steam oxidation data, a test program was proposed (OHRD Report M87-73-K) to carry out the appropriate oxidation experiments in a steam environment. This program was approved and supported by the Nuclear Studies and Safety Department of the Design and Development - Generation Division.

To facilitate the above-mentioned testing, the existing Fuel Cladding Oxidation Test Facility (FCOTF) was modified to permit oxidiation experiments to be carried out in a steam environment. The main components of the newly developed steam loop are: (a) a custom-built demineralized feedwater system; (b) a laboratory-type mantle heater/glass vessel steam generator; (c) a custom-built cylindrical stainless steel vessel or reaction chamber; (d) a water-cooled condenser system. A schematic diagram of the newly built steam loop is shown in Figure 8.

Results obtained from steam oxidation tests carried out at 600°C indicate that both parent and braze HAZ cladding materials oxidize at measurably lower rates than those observed in an air environment. However, the results also indicate that the remaining substrate metal in highly oxidized samples was more brittle than similar materials oxidized in air. This apparent additional deterioration of the substrate metal is suspected to be caused by absorption of hydrogen produced by the Zr metal-steam reaction. Analysis of steam oxidation experiments in progress or scheduled should help quantify the above preliminary observations.

3.5 General Materials Testing in the High Temperature Deformation Laboratory
(E.T.C. Ho and M. McGraw)

In the past year, the capability of the computer-controlled Materials Characterization Test Facility (MCTF) was augmented by the acquisition of a high temperature extensometer. This extensometer, together with the split three-zone resistance furnace, has enabled us to carry out tensile testing of a wide range of materials and specimen geometries. In addition, several new application programs for test control, data acquisition, data analysis and data plotting were developed for use with the computer-controlled MCTF.


REGULATING VALVE
SHUTOFF VALVE
TAP WATER
CRYSTALAB DEEM-A-FLOW IN-LINE DEMINERALIZER (MODEL DF-500)
FLOAT ACTIVATED SOLENOID VALVE SWITCHING UNIT
INDUSTRIAL IN-LINE 5 µm FILTER AND HOUSING
SCH 40 STAINLESS STEEL PIPE REACTION VESSEL
ZIRCALOY-4 SPECIMEN
SUPERHEATER
SAMPLE FURNACE
SPECIMEN FLANGE ASSEMBLY
EXISTING IN-HOUSE BUILT TUBE RESISTANCE FURNACES
TO DRAIN
STAINLESS STEEL COVER
STAINLESS STEEL CORE COPPER JACKET CONDENSER
4000 ml FLASK
STAINLESS STEEL SLOSH SHIELD
TAP WATER
TO DRAIN
① MODEL LV-40 316SS HIGH TEMPERATURE LIQUID LEVEL SWITCH (PROVIDE SIGNAL FOR FLOAT ACTIVATED SOLENOID VALVE)
LITTLE GIANT CONDENSATE REMOVAL UNIT

Schematic diagram illustrating the layout of the main components of the steam loop custom-built for the Fuel Cladding Oxidation Test Facility.

FIGURE 8

A number of requests for tensile testing to obtain detailed true stress-strain curves were processed using the above test facility. Materials tested ranged from carbon steels to Zr-2.5 wt% Nb pressure tube materials.

3.6 Materials Properties for CANDU Reactor Ex-Core Component Alloys

(E.T.C. Ho and M. McGraw)

3.6.1 Materials Properties for Darlington PHT Piping Material

Data obtained from a materials test program carried out as part of the Darlington leak-before-break study showed that some non-stress relieved submerged arc weld (SAW) metal exhibited lower than expected fracture toughness at 250°C. The above test program also showed that a post-weld heat treatment (PWHT) of two hours at 620°C improved the 250°C fracture toughness and the room temperature Charpy impact energy of the weld metal. In light of the above findings, Task Group B-5 recommended that further investigation of the effects of PWHT on the SAW metal be carried out before PWHT is recommended for the 22" lines of Darlington Units 3 and 4.

The Materials Integrity Section responsibility in the above materials evaluation program was to characterize the tensile plastic flow behaviour of both as-welded (AW) and PWHT PN-229 weld materials over the temperature range 20-300°C. At each temperature, the specimens were deformed under constant applied strain rates of $1x10^{-4}$, $1x10^{-3}$, and $1x10^{-2}$ s^{-1}.

Results obtained indicate that both AW and PWHT materials were susceptible to jerky or serrated flow. However, the AW material was much more susceptible to this dynamic strain aging phenomenon, especially over the temperature range 150-200°C. The data also showed that the AW material exhibits a peak in tensile strength at about 200°C and a trough in ductility at about the same temperature. These findings indicate that the AW material should exhibit minimum fracture toughness properties at about 200°C. Furthermore, the results indicate that although the PWHT used did not completely suppress serrated flow, it apparently eliminated sufficient amounts of interstitial solutes from solid solution to improve the ductility of the PWHT material significantly. This improvement in ductility coupled with the reduction in tensile strength should give the PWHT material much better fracture toughness properties than the AW material over the temperature range 150-200°C. A detailed description of the above test results is given in OHRD Report 87-319-K.

3.7 Observations of the Bainite/Austenite Interface in Steels

(V. Perovic)

Some of the results on the Bainite/Austenite interface in steels were presented at the international conference on phase transformations in solids, University of Cambridge, England, July, 1987. The study is collaborative work with Prof. G.R. Purdy, McMaster University. Transmission electron microscopy was used to examine in detail the structure of the bainitic ferrite/retained austenite interface in series of steels, including Fe-C-Ni and Fe-C-Mo-Si alloys. The general aspects of the interfacial structure include sets of parallel interfacial steps each of which exhibits a pronounced strain field contrast, and weakly-imaged sets of interfacial dislocations, spaced 1 to 3 nm apart. The interpretation of the phase transformation in steels is of benefit to our Zr-Nb studies where the $\beta_{Zr} \rightarrow \alpha_{Zr}$ phase transformation is important from deformation and hydride precipitation aspects.

4.0 FRACTURE MECHANISMS UNIT - 1987 REVIEW

4.1 Blisters
(M. Leger and O. Kiir)

4.1.1 Blister Fracture

This program was carried out in cooperation with A.C. Wallace of the NDFE section. The aim of the work this year was to carry out parametric studies of the variables thought likely to influence the stress required to fracture hydride blisters in Zr-2.5 wt% Nb pressure tube material. The program was initiated at the end of 1986 and was in progress throughout 1987. It was funded by NSD through COG.

The tests involved growing blisters (by air jet technique) under different conditions of temperature, jet diameter, stress, thermal cycling, dissolution and surface geometry and then testing the resulting blisters to fracture in four point bend tests either at room temperature or at elevated temperature. One of the proposed tests, growing blisters under stress to fracture, was not started during this year. Approximately eighty blisters were tested to fracture. The results to the end of the year basically support previous observations that the major parameter influencing blister fracture is the depth of the blister.

Growth temperature (bath temperatures of 300 or 350°C) had little effect on the measured fracture stress.

Blister diameter does have an effect with smaller diameter blisters having apparently smaller fracture stresses. However, the effects of hydride distribution within blisters of different diameters has not yet been taken into account.

Blisters grown under stress behaved significantly different from those grown without stress. Blisters grown under compressive bending stresses cracked at lower applied stresses than those grown under tensile bending stresses. It is not yet clear whether these effects are due to the presence of residual stresses left in the samples following their preparation or if it is a real effect of the stresses applied during blister growth.

No thermal fatigue crack initiation occurred in blisters which were thermally cycled 100,000 times by turning the air jets on and off. The blisters were subsequently tested and the measured fracture stresses were within the normally observed range for blisters grown under steady conditions.

Blisters have now been grown on notched specimens and will be tested in the coming year.

Initial statistical analysis of the results has been completed and a report is in preparation.

A major part of the blister fracture program has been the characterization of the blisters themselves. Since the blisters are a distribution of hydride rather than a volume of solid hydride, the problem of characterization by a few parameters is not a trivial one. We also inherently believe that the distribution of hydride will have a significant influence on the fracture stress. Two different approaches have been applied: 1.) Direct measurement of the deuterium profiles in blisters grown with deuterided samples and 2.) Hardness profiles on cross-sections through the

blisters. The deuterium profile measurements were carried out at Queen's University on a contract basis. During the year, measurements were made of the deuterium profiles through thermally cycled blisters and several methods were used to check the calibration of the measurements in absolute terms. Analysis of the results has not yet begun, but these should provide a very severe test of the blister growth model. Hardness profiles have been carried out on many different blisters. An initial comparison of the hardness profiles with the deuterium profiles taken on the same set of blisters indicated that there was a good correlation between the two.

Work in progress at WNRE in the blister fracture area has been periodically reviewed. Blister growth techniques at WNRE are different from those at OHRD and the blister shapes have also been significantly different. There appears to be more scatter in the blister fracture stresses in the WNRE data than in the OHRD data but a similar trend of reduction in blister fracture stress with blister depth is apparent. The stresses at which blisters can initiate DHC cracks are in general agreement with the blister depth effect noted above. A critical review of a WNRE paper prepared for publication was carried out and significant changes were recommended.

4.1.2 Development of the Probabilistic Model of Reactor Assessment for Blisters

(M. Leger and D. Leemans)

This work was underway throughout the year and significant progress was achieved. It has been a cooperative effort with Dr. D.V. Leemans of Crosscurrents Research and Policy Consulting taking the lead role in the computer model development with significant inputs from Dr. T.P. Byrne of the Applied Mechanics section (blister growth modelling) and from M. Leger for aid in incorporation of other aspects of the TRAC work.

The main objective of the work is to produce a probabilistic assessment of pressure tube rupture due to blister formation at pressure tube calandria tube contacts for a complete reactor. At the end of 1986, a model had been developed which could be used to assess the probability of having blisters in a reactor. The output of this model was a significant factor in the decision to remove a pressure tube from Pickering 3 during 1987 since it showed the extreme sensitivity of the assessment to the deuterium ingress rate. In 1987, the main objective was to incorporate blister growth into the model. This necessitated reworking the model to a significant extent. A great deal of effort was expended on generating and reviewing blister growth model results. With the surprising deuterium levels observed in the tube actually removed from P3, there was a need for the best assessment with the model in its current state. A rapid response to the need was achieved by taking some shortcuts in the blister growth modelling and by assuming that there was a critical blister size that would lead to a DHC crack under reactor operating pressure.

The model, as it now stands, produces an assessment of the probable number of tubes with critically sized blisters as a function of reactor operating time. It takes into account the garterspring distribution, the contact times for different spanlengths, the spreading of the contact with time (provided by R.A. Holt), the deuterium ingress rates, the uncertainty in the critical blister size based on the results of the blister fracture program and the blister growth rates derived from the blister growth model. The model is currently being applied to an assessment of P4 following the measurements of deuterium in 20 channels and is forming a major part of the argument to the AECB for the restart of the reactor.

An external contract with NBEPC was successfully completed. It involved carrying out an assessment of blister formation in the Point Lepreau reactor using the first version of the model.

4.2 Delayed Hydride Crack Initiation
(W.K. Lee and P. Vesely)

4.2.1 Rolled Joint Integrity

This test program was designed to examine the performance of zero clearance rolled joints with machined defects under simulated reactor operating condition. The objective was to compare the test results with those obtained from small specimen laboratory tests.

Defects were machined into the inside surface of the pressure tube and then a diffusion anneal was carried out at 300°C without applied stress to produce about 60 ppm hydrogen in the joint and the body of the pressure tube. The EDM notches were machined into the ID surfaces near the burnish marks where the residual stresses are highest and at the middle of the pressure tube. The notch profiles were measured by the plastic replication method. With a test pressure of 10 MPa and the residual stress present in the joint, the highest pseudo stress intensity at the most severe notch was estimated to be about 7.8 MPa $m^{0.5}$.

The furnace and pressurization equipment for service simulation were assembled, however, problems with the threaded plugs and seal weld of rolled joint assembly resulted in test delay.

4.2.2 Load Maneuvering

This program addressed the concerns about the effect of load following operation on crack initiation. A matrix scoping test using the tensile type of specimen exposed to cyclic thermal and mechanical loading was formulated to investigate the worst case situation.

The notched tensile specimens were prepared from flattened and hydrided Zr 2.5 wt% Nb pressure tube. The hydrogen concentration was about 80 ppm, well above the TSS for dissolution at the test temperature of 250°C. The hole drilling technique was utilized to determine the amount of residual stress after the straightening and hydrogen diffusion steps. The highest tensile residual stress was about 100 MPa and the compressive stress was about 100 MPa. The typical specimen contained a U shaped notch with root radius of 0.2 mm and a depth of 0.8 mm. The elastic stress field at and near the notch position was estimated using the standard closed form equation based on Linear Elastic Fracture Mechanics. This stress profile was comparable to that obtained by the finite element method. A plane strain condition was found to exist near the notch root.

Major components for the cyclic test facility were received, assembled and commissioned. The three zone furnace was connected to the three temperature controllers and mounted onto the INSTRON tensile machine. The temperature profile was measured for a tensile specimen with a gauge length of 25 mm, the maximum temperature variations in the specimen were 2°C at 250°C and 1°C at 300°C. The data acquisition system to record the test temperature and to determine the onset of crack initiation included a constant current supply for the dc potential drop method and the HP85B-HP3421A computer based data logger.

4.2.3 Hydrogen Diffusion in P8 Fretted Tubes

The experimental program to demonstrate that the hydrogen concentrated at the fretted area will diffuse away under simulated reactor operating conditions was successfully completed. The 60 day and the 220 day diffusion anneal test results showed that the hydrogen had diffused away from the fret mark produced by the dummy fuel bundles used in the hot conditioning of P8 pressure tubes. There was no tendency for the secondary pit-like marks to prevent the hydrogen from diffusing away and no crack initiation had occurred.

The experimental data showed that the rate of hydrogen removal from the fret mark was faster than the theoretical prediction. The theoretical prediction was based on the worst case assumptions which included:

(1) The average initial hydrogen concentration under the fret mark is the maximum level associated with the deepest part of the fret mark. This is unlikely to be the real case as the fret mark depth varies significantly and these should be correlated to changes in hydrogen concentration.

(2) The hydrogen diffusion calculation is one dimensional, the circumferential hydrogen diffusion contribution was not included in the prediction. Therefore, it is expected that the theoretical predictions will lag the actual rate of hydrogen redistribution.

The above experimental results supported the calculations used in the original hydrogen diffusion evaluation for the fretted tubes in P8. The Atomic Energy Control Board, based on this information, accepted Ontario Hydro's request to defer the tube removal specified in the original license condition. This tube removal for hydrogen determination was to answer the concern about the validity of the theoretical predictions.

This experimental program has resulted in significant saving to the corporation since the cost of removing an active pressure tube and replacement energy costs are very high.

4.2.4 Crack Initiation in Pressure Tube Materials

The main objective was to determine the crack initiation criteria under increasing hydrogen and end-of-life service conditions. Crack initiation experiments using the four point bend specimens under deflection control were carried out and analysis of the stress state leading to crack nucleation was also attempted.

Failures had occurred in four point bend specimens with notches on the ID surface after exposure to thermal cycling under stress. A typical thermal cycle involved heat up to 300°C and then holding at 250°C. Crack nucleation was found at the root of a V shaped microdrilled notch after 43 thermal cycles. The specimen was loaded to 500 MPa and the defect was about 0.8 mm deep and 5 mm long. Another specimen with an edge to edge notch 0.7 mm deep also failed after 182 thermal cycles. The applied stress was 400 MPa. Both specimens contained multiple crack initiation sites.

Failures had also occurred in those four point bend specimens with a thermal gradient to simulate the continuous hydrogen ingress condition. The portion of the specimen under constant applied stress was kept at 250°C and a temperature gradient of 30°C was used to promote hydrogen diffusion from the hydride layer at the specimen's edge to the stressed area. The specimens with

an initial 30 ppm hydrogen were subjected to one thermal cycle prior to the temperature gradient test. A one-dimensional hydrogen diffusion calculation indicated that new hydride precipitation would occur after six months (4320 hours) based on the 2 TSS model. Two duplicate specimens with 0.7 mm deep notch and loaded to 400 MPa failed in about the same time (5000 hours). Multiple crack initiation sites were found on the fracture surfaces. At 2800 hours, these two specimens showed only very small amount of hydride reorientation at the notch tip. The failure time and the hydride morphology observed compared very favorably with the theoretical prediction of time to hydride precipitation.

Comparison between the thermally cycled specimens and the hydrogen ingress specimens indicated that although the hydride precipitation methods were different, similar crack initiation times were obtained. The crack initiation times were also of the same order of magnitude as those in past cantilever beam specimens exposed to thermal cycling conditions. The axial/hoop biaxial stress state in a four point bend specimen was similar to that in a pressure tube, whereas the cantilever beam specimen only had the hoop stress component. The larger specimen size also provided more potential nucleation sites than the cantilever beam specimen.

By comparing the stress state under a notch to the threshold stress intensity value for pressure tube material, a boundary curve for nondamaging notches was generated. The stress intensity factors (K, mode I) for each combination of applied stress, notch depth and root radius were determined based on Lukas's work. Cracking will not occur if the calculated stress intensity factor is less than the threshold stress intensity factor. This approach is conservative since a small crack (of the order of the hydride size) is assumed to be present ahead of the notch. A boundary curve of applied stress versus notch depth for the nondamaging notch can then be derived for each set of notch root radius and threshold stress intensity value to satisfy the above condition.

The experimental results from failed cantilever beam specimens (from AECL-RC) were found to be above the boundary curves determined by the above method. This comparison was qualitative since details about the notch geometry and the test setup procedures were not available. The advantage of this approach is that we can analyse the severity of a defect without relying too heavily on experimental data which can only be obtained in long-term tests.

The probability of crack initiation was estimated by deriving the probability value from an assumed probability density function that was dependent on the stress state in the material. Using a stress intensity dependent probability density function and an assumed rate of failure, the mean time to crack initiation was obtained. This function matched qualitatively with the experimental cantilever beam test results for Zircaloy-2. There is limited Zr-2.5 wt% Nb data for comparison.

The above analysis provided a method to project into long test time to determine the probability of crack initiation. A preliminary sensitivity analysis of the probability function was carried out to determine the effect of variation in the input assumptions. With the threshold stress intensity as a distributed value, upper and lower bound curves were produced for the times to failure as a function of the applied stress intensity. The cantilever beam experimental data were found to lie between these two boundaries.

4.2.5 DHC Crack Initiation from Cracked Blister

To determine if delayed hydride cracking will start from cracked blister, four point bend crack nucleation tests were set up with potential drop monitoring to detect the onset of cracking. The

standard four point bend loading fixture was modified to apply a constant hoop stress on the blister at the OD surface. Room temperature strain gauging indicated that the experimental loading parameters were very close to the theoretical values based on beam theory. Four specimens are being tested and the applied stress intensities at the blister crack tip were equivalent to those in the reactor. No indication of cracking was observed after about 2 weeks of test at 250°C.

4.3 Delayed Hydride Crack Growth Studies
(G. Shek and H. Seahra)

This program is funded by the CANDU Owners Group (COG) under Work Package #6505. The objectives of this research program are:

(i) to evaluate the effects of metallurgical (hydride morphology, strength, and manufacturing processes) and operating (thermal and loading history, hydrogen ingress) parameters on delayed hydride crack velocity;

(ii) to provide more crack growth data on pressure tube material at operating temperature and under thermal cycles.

In 1987, the computerized data acquisition and control system was upgraded with the replacement of the HP85B computer with a HP9816. The BASIC program was up-loaded and modified for more frequent data logging and better data analysis. Improvments were made to the DHC test rigs. These included the addition of a load cell and modification of the specimen holders and furnace to obtain more uniform specimen temperature.

The following summarizes the test results:

Strength: More tests were performed confirming earlier results that DHC velocity in pressure tubes is strength dependent (Figure 9). The fracture surface displayed narrower striation spacings at higher strength level (Figure 10). Figure 10 also indicates that the spacing is narrower towards the end of the crack where the K is higher. The different strength levels of the specimens were obtained either by high temperature stress relief or by thermomechanical treatments (TG3 hold point materials). In order to have more systematic variation of strength, Zr-2.5 wt% Nb extrusions were cold rolled to different amounts of cold work up to 60%. This group of specimens is now being tested.

The tests also indicate that there may be a threshold strength level below which DHC will not readily occur. The threshold level in pressure tube can be obtained either by increasing the test temperature or by metallurgical treatments (eg. high temperature stress relief). It is difficult to initiate DHC at the outlet end temperature (290°C) in pressure tubes containing 100 ppm hydrogen and cracking could not be initiated at 250°C in a sample weakened by a 600°C stress relief.

Hydride Morphology: Previous tests have shown that hydride morphology does not have a significant effect on the stage II crack velocity obtained by approaching the test temperature from above. In these tests, new hydrides were formed as a result of the thermal cycle. Further tests were performed at 180°C with the test temperature approached from below to avoid the

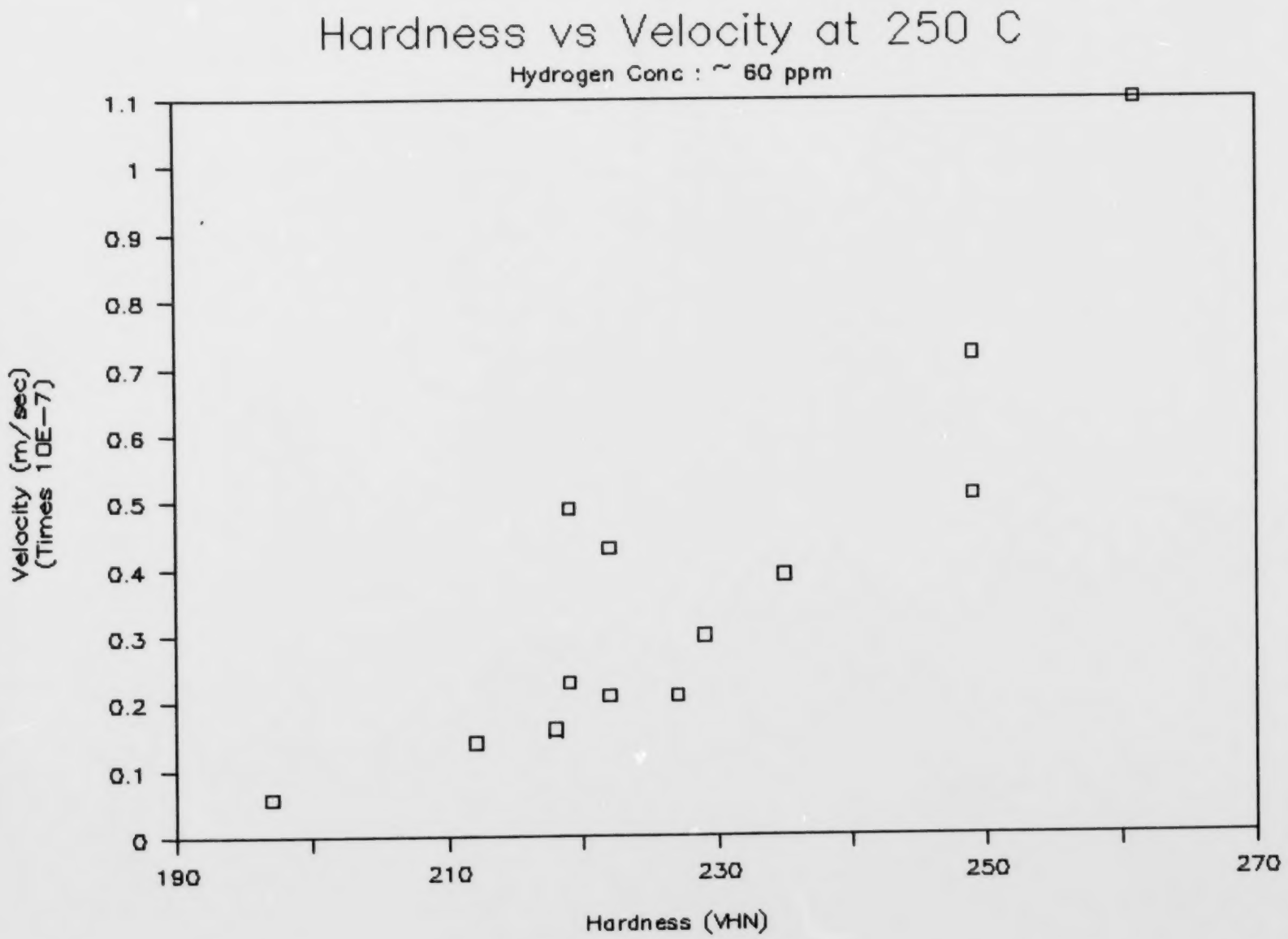

Hardness vs Velocity at 250 C
Hydrogen Conc : ~ 60 ppm
Velocity (m/sec)
(Times 10E-7)
1.1
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
190
210
230
250
270
Hardness (VHN)

FIGURE 9

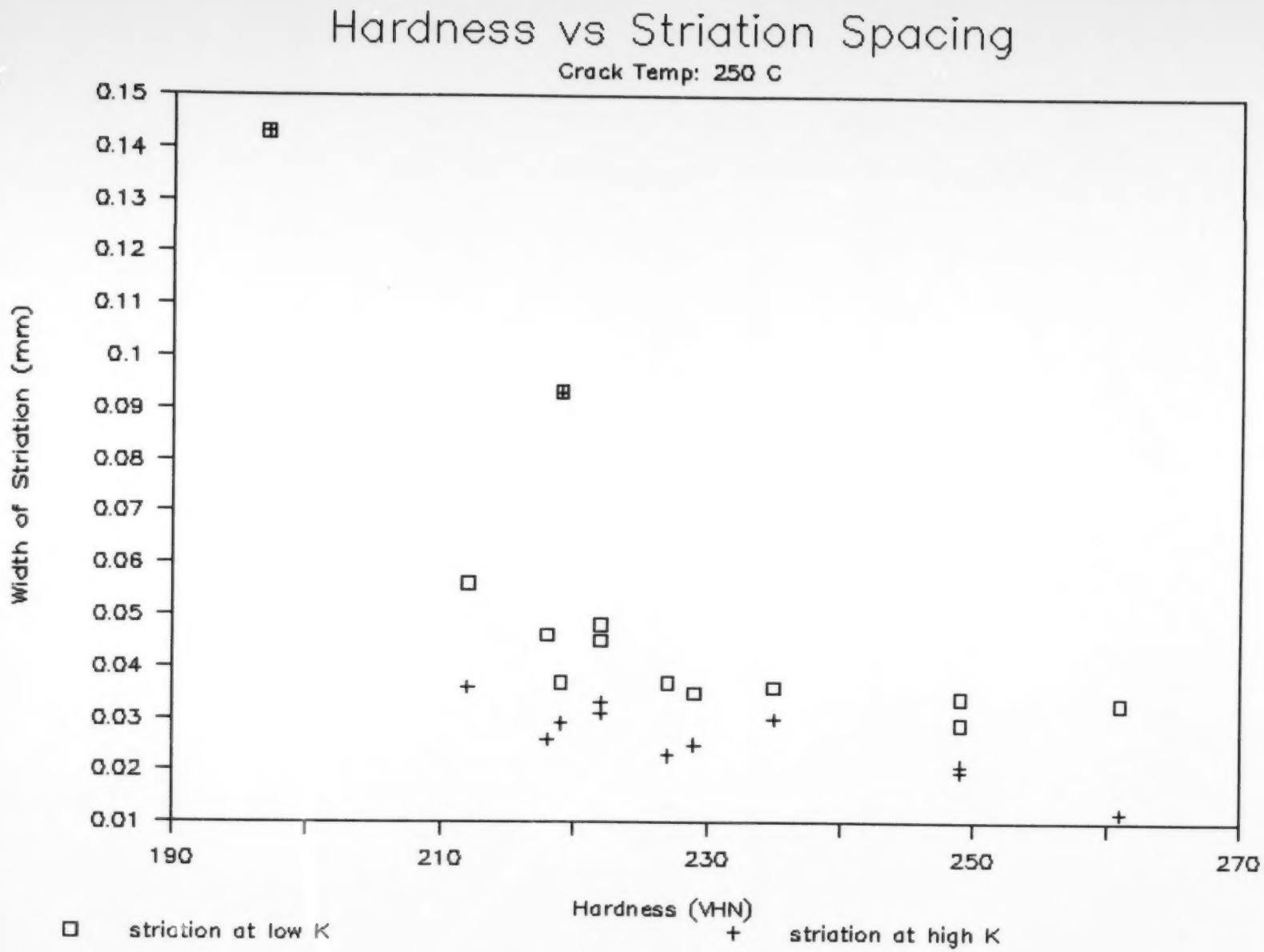

Hardness vs Striation Spacing
Crack Temp: 250 C
Width of Striation (mm)
0.15
0.14
0.13
0.12
0.11
0.1
0.09
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
190
210
230
250
270
Hardness (VHN)
striation at low K
striation at high K

FIGURE 10

forming of new hydrides in the matrix. The results show that samples containing fine radial hydrides crack about twice as fast as those containing coarse circumferential hydrides under this test condition.

K (Stress Intensity) Maneuver: This part of the work program was partly funded by CNS. Constant K specimens were used to examine the effects of K maneuver on DHC behaviour. The results showed that an incubation time was required to resume cracking after the applied K of an active delayed hydride crack was significantly reduced. The length of incubation period was delta K dependent and reached 10-15 hours for a delta K of 10 MPa.$m^{1/2}$. Test results also show that during the incubation period after K reduction, lowering the test temperature was not effective in reducing the incubation time. However, cracking resumed immediately when the K is returned to the original high level.

Crack tunneling was observed in most of the specimens tested. The extent of tunneling (i.e. depth of ligament) appears to increase with decrease in strength either through temperature or metallurgical effects.

4.3.1 Flow Injection

The flow injection refuelling system to be installed at BNGS-A will lower the inlet end temperature by 10-15°C for a short period of time. There is concern about occurrence of DHC as a result of the temperature drop in conditions that are conducive to DHC (presence of defects, high stresses and hydrides). A test was performed using a 60 ppm constant K specimen to address this concern. The test shows that DHC occurred with a minimum temperature drop of 13°C from 250°C, but at a considerably lower velocity than the published crack velocity data. There was an incubation period of 1.5-2 hours before crack initiation. Cracking stopped when the temperature was restored to 250°C. Based on the results of this demonstration test, it is unlikely that any significant crack extension will occur as a result of flow injection.

4.4 Effects of Hydride Morphology on Fracture Toughness

(G. Shek and H. Seahra)

This is a co-operative program with the Fracture Mechanic Unit to determine the effects of hydride morphology on fracture toughness. Previous attempts to produce mixed hydrides in strips straightened from pressure tube rings have only had limited success because of residual stresses from straightening. It was decided that the best solution to this is to hydride the tube under internal pressure to avoid the problem of residual stresses from sample preparation. Small curved compact tension specimens will then be used to determine the effects of temperature and hydride morphology on fracture toughness.

A procedure was developed to produce mixed hydrides in which the hydrogen was diffused into the tube from a surface hydride layer and the hydrides were precipitated under intermediate pressure (12.4-13.5 MPa). Titanium end plugs were electron beam welded to each end of the tube to form the pressure boundary. The hydriding under internal pressure was performed in a pressurized water loop and the specimen was heated by a three zone furnace.

Three tube samples were hydrided to nominal concentrations of 40, 60 and 80 ppm. The hydrides had a mixed orientation similar to that observed in the Bruce Unit 2 X-14 rolled joint region. The 40 and 60 ppm samples have been tested by the Fracture Mechanics Unit. The mixed

hydrides had a detrimental effect on fracture toughness for test temperature below ~150°C. The mixed hydrides at these two concentration had no effects on the fracture behaviour at 175°C and above.

4.5 Effect of Residual Grain-Interaction Stresses on Hydride Morphology in Zirconium Alloys

(V. Perovic and R. Jarochowicz)

The research on the role of internal stresses on hydride precipitation in coarse-grained Zircaloy-2 swaged rod continued. Previous experiments have demonstrated that the residual grain-interaction stresses produced as the consequence of anisotropic thermal expansion or uniaxial deformation can effect the way in which hydride precipitate. Slow cooling after annealing at elevated temperature leaves the basal planes in tension (regardless of the previous thermo-mechanical history) and the hydrides grow transgranularly. Tensile deformation of the sample produces compressive residual stresses normal to the basal planes and the hydrides lie along the α/α boundaries, except for grains whose <c> direction is more inclined towards the rod (tensile) axis, where the transgranular hydrides are observed. The calculated and measured internal stresses have shown that the basal planes in those grains are left in residual tension. Compressive deformation of the sample increases the tensile residual stresses normal to the basal planes, but in addition introduces deformation twins. Hydrides can be found either along α/α, twin boundaries or transgranularly, depending on annealing treatment after deformation. The role of twin boundaries is not clear. In order to get more insight into the role of twinning on residual internal stresses and consequently on hydride precipitation a more compressive program has been initiated. The objective of this program is twofold;

i) to measure residual grain-interaction stresses and characterize hydride distribution in the compressive specimens deformed at lower strains (0.5-2%); in this deformation range twinning is not observed.

ii) to estimate residual stresses in the high temperature (600°C) annealed specimens previously compressed 4%. It is shown that this heat treatment had recovered dislocation substructure without removing twins introduced during compression. Under those conditions twin interfaces are free of hydrides and only transgranular hydrides are observed.

4.6 Offcut Examination

(G. Shek)

We continue to be involved in the evaluation of pressure tube offcuts which have UT indications. The majority of the work was sub-contracted to AECL-CO. Metallographic examination did not detect any B2N06 type defects in the offcuts. The UT indications were mostly from surface imperfections as a result of the sandblasting and from shallow surface drawing marks and scratches.

A crack-like defect in the axial-circumferential plane associated with stamped marks on a Pickering unit 7 offcut (D1011) was determined to be a delayed hydride crack. The tube was stamped and stored for more than two years before autoclaving. Cracking was initiated as a result of the high residual stress and the stress concentration associated with the stamped letter "1". The crack grew for 3.5 mm during the storage period and eventually stopped after the residual stress in the tube was relaxed by the autoclave treatment.

Zirconium carbides were found in some offcuts which have UT indications. The inclusions are cracked, but the cracks do not extend into the metal matrix. The carbide inclusions were only detected in offcuts of pressure tubes made from ingots containing high carbon content. The specification of ingots was subsequently revised to lower the carbon content.

4.7 **Hydriding Services**
(G. Shek, P. Vesely and H. Seahra)

A number of pressure tube samples were hydrided to support programs in other sections: the NDE of hydride platelets in pressure tubes; corrosion fatique of hydrided Zr-2.5 Nb pressure tubes and the electromagnetic studies of hydrides.

4.8 **Development of a Technique for the Nondestructive Evaluation of Hydrogen in Zirconium Pressure Tube Material**
(K. Tashiro & D. Oad)

The major goal of 1987 was the measurement of the effects of hydrogen upon the resistivity properties of Zr-2.5 wt% Nb. Initial measurements were performed on small samples. Small samples were instrumented and holding blocks were constructed for controlled heating and inert gas cooling. This experimental setup allowed measurement of the effect of hydrogen on both resistivity and thermoelectric power of Zr-2.5 wt% Nb.

Measurements of thermoelectric power on an 80 ppm hydrogen sample indicated that the discontinuity seen at the correct temperature was characteristic of the zirconium itself and was not hydride related. This result would confuse the resistivity measurements, if the dc potential drop technique was utilized. In order to prevent this effect from occurring, the ac potential drop technique of resistivity measurement was employed. Thermoelectric power measurements as a method of hydrogen determination was abandoned at this time.

New resistivity measurements of 14 ppm and 72 ppm hydrogen samples were very encouraging. It appears that 14 ppm was a reasonable lower detection limit. At this point, hydrogen concentration can be derived from the TSS transition temperature and also from the integrated peak area of the first derivative of the resistance curve with respect to temperature.

Due to the deuterium results of P3 L09, an accelerated test program had to be implemented. Based on the favourable results of the small scale test results from 14 and 72 ppm hydrogen samples, the decision to proceed was made and the construction of a large scale (actual tubes) test facility initiated.

Preliminary tests were performed on sections of actual nonirradiated pressure tubes with hydrogen concentrations of approximately 40 and 14 ppm hydrogen. Contacts were spot welded and heating was accomplished by wrapping a ceramic heater band on the outside of the tube. Electromagnetic induction was compensated by introducing a separate sense loop for the induced signal only. The TSS_d and TSS_p peaks in the temperature derivative of resistivity have been found for each. However, the signal quality of these result are poor with a large amount of noise being present in all measurements. This would make quantitation of the measurements difficult. The signal quality could be improved with a more sophisticated signal conditioning system.

4.9 EPRI Project RP1890-5: Corrosion Fatigue Boiler Tube Failures in Waterwall and Economizers

(E.T.C. Ho and R.E. Irwin)

Although official approval for this EPRI-funded project was not received from New Business Ventures (NBV) until December, we proceeded with the project throughout the year. Significant progress was made in the past year. The project accomplishments are best described under the two headings "Flow Loop and Test Facility Developments" and "Test Procedure and Software Developments".

4.9.1 Flow Loop and Test Facility Developments

A review of the original design for the Boiler Tube Test Facility (BTTF) indicated that several modifications should be implemented. The more significant modifications made to the BTTF were as follows:

1. replacing the plastic feedwater tank with a stainless steel design;
2. replacing the recirculating pump which has a plastic housing with a stainless steel recirculating pump;
3. upgrading the accumulator for the pressurizing circuit from 3500-5000 psi;
4. adding a 1/2" wall carbon steel cylindrical vessel for safety containment of the tube specimens which are subjected to high temperature and high pressure water under test conditions.

Several hardware components and computer equipment were acquired to increase the capability of the BTTF. Computer equipment acquired include a Toshiba model P-351 printer and a Hewlett-Packard model 7475A, 6-pen digital plotter. Hardware components acquired include a new microprocessor-based temperature controller for the preheater and a strain gauge assembly and associated power supply for the HP model 3497A Data Acquisition System.

A new test rig capable of producing cyclic loading in plain tube specimens was designed and built. The main reason behind selecting a plain tube geometry over a welded-fin geometry is that the loading in the former is deemed more reproducible. The use of plain tube specimens should result in the experimental data being more consistent and easier to analyze.

4.9.2 Test Procedure and Software Developments

An initial design for the tube specimens was selected and the rationale for choosing a plain tube geometry has been mentioned above. To date, two tube samples have been fabricated and they are now ready for testing under high temperature and high pressure conditions. Two twenty feet lengths of 2" ID x 1/4" wall SA210A1 boiler tube material were obtained from Dr. M.T. Miglin of B&W. These boiler tubes were purchased by B&W specifically for the EPRI project. The use of this material for our test program should give our data better correlation with the small specimen data to be generated by B&W in their autoclave tests.

During the past year, several computer programs were developed for test control, data acquisition and data analysis using the HP VECTRA personal computer. This type of software development is expected to continue in 1988.

Analysis of test data obtained from cyclic compression loading of a strain-gauged plain tube specimen enabled us to develop a test procedure whereby ID-initiated cracks can be generated consistently in a plain boiler tube specimen. This test procedure will be further refined before it can be applied to the experimental tube test program for the project.

5.0 1988 PROGRAMS - DEFORMATION MECHANISMS UNIT

5.1 Reactor Data Analysis

5.1.1 Assessment of Power Reactor Operating and Deformation Database

The existing database of power reactor deformation and operating data will be reviewed, including elongation, diametral and sag data, contact data and fast flux and temperature profiles. Deficiencies in the database will be identified, and work started on upgrading and maintaining it in a comprehensive and easily accessible form. Plant measurements of pressure tube deformation and pressure tube/calandria tube contact will be added to the database as they become available. The crystallographic texture of offcuts from fast and slow elongating B2 pressure tubes will be examined to determine if there is a correlation with elongation rate.

5.1.2 Fuel Channel Sag

A new version of CDEPTH will be completed in cooperation with the Applied Mechanics Section. This version will include an unloading-reloading feature to allow for the difference between operating and measuring conditions in a consistent manner, and will use simpler beam elements which will speed up the computations without a significant loss of accuracy. Analysis of existing sag and contact data will continue. Modifications to CDEPTH inputs will be made reflecting the review of operating and measuring conditions. New data will be analyzed as it becomes available. Evaluation of the pressure tube design equations will be performed in the manner described in Section 3.1.1, and, in cooperation with AECL-RC, the calandria tube design equations will be modified to conform to sag deflection data.

5.1.3 Contact Spread Modelling

The study of contact development will continue and an updated version of the contact spreading model will be derived. A study of the sensitivity of contact spreading to factors such as temperature and flux profile and initial pressure tube/calandria tube gap will be carried out. As a result, parameters for the model will be derived for each reactor based on its operating conditions.

5.1.4 Pressure Tube Elongation

The crystallographic texture of the offcuts of fast and slow elongating Bruce Unit 2 pressure tubes will be measured and correlated with elongation rate.

5.2 In-Reactor Deformation Mechanisms

5.2.1 DIDO Irradiation Experiment

The experiment will continue throughout 1988. A fast neutron fluence of 8.8×10^{25} n/m^2, E>1 MeV will be achieved in the most highly irradiated specimen (the axially averaged fluence in a central P3 channel is currently about 7.6×10^{25} n/m^2, E>1 MeV.

Growth results will be obtained from specimens of surveillance tubes P4 K10 and B2 P12 which were introduced to the program in November 1987. These results will reveal whether the presence of stress during irradiation has affected the growth rate in comparison to specimens irradiated for zero fluence with no stress.

The thin foil blanks made from growth specimens removed from the program will be shipped to OHRD for examination. The results will be compared to those obtained from the surveillance tube specimens and will assist in understanding the discrepancy between design equations and the Dido growth results.

The interpretation of $^{93}Nb^{m}$ activation results from growth and surveillance tube specimens will be completed giving a direct comparison of the fast flux (>0.5 MeV) in Pickering, Bruce and Dido.

A preliminary assessment will be made of the effect of temperature on the steady-state growth rate of standard and modified pressure tube materials.

5.2.2 Mechanisms of Growth

The theoretical model for the microstructure dependence of growth will be further developed with better approximations of the terms describing the grain boundary sink strengths. The predictions of steady-state growth rate will be assessed.

5.2.3 End-of-Life Deformation Characteristics

A contract will be placed with the CEA to irradiate 6 creep and 36 growth specimens in one of the high flux reactors in France at a temperature of 280°C at a fast flux of 2×10^{18} n/m^2/s, E>1 MeV. The target irradiation dose will be 1.5×10^{26} n/m^2, E>1 MeV (axially averaged lifetime fluence for Pickering A, with the option to extend the fluence to 3.0×10^{26} n/m^2, E>1 MeV (peak lifetime fluence in Darlington). Irradiation is expected to start in late fall.

5.2.4 Thermal Creep

The study of anisotropy and stress dependence of thermal creep, and its dependence on position along the pressure tube will be continued. Biaxial tests at 300°C will be initiated to assess the effect of position along a pressure tube on the creep rate. An experiment to assess the effects of thermal cycling on the creep rate of pressure tubing under biaxial stress will also be initiated. The results of these experiments will assist in defining more accurately some poorly understood aspects of the in-reactor deformation of pressure tubes.

5.2.5 TEM Studies of Irradiated Zr-Nb Alloys

The microstructures of specimens from surveillance tubes and irradiation growth specimens irradiated to high fluences will be evaluated and compared with preirradiation microstructures. The results will be related to irradiation induced changes in properties.

As reported the dislocation substructure in the unirradiated Zr-Nb pressure tubes is composed of dislocations with different Burgers vectors. The presence of different types of dislocations and their spatial distribution should influence irradiation growth by altering the manner in which irradiation produced defects are partitioned among various sinks. The high fast neutron fluences is expected to have an effect on phase stability and may also cause microchemical changes. Thus, the TEM studies should provide us with very valuable information on the effect of neutron irradiation on dislocation substructure, phase stability and the nature of neutron damage in the pressure tube materials. Access to an analytical transmission electron microscope will be required to perform these studies. The current TEM at OHRD is 15 years old and has very limited capabilities.

Further work will also involve TEM examination of radiation damage generated in creep and growth specimens irradiated in the proton accelerator. The results of these analysis will be used in a study of the mechanisms contributing to creep and growth in zirconium alloys.

5.3 Pressure Tube Development

5.3.1 Assessment of Production Task Group 3 Tubes

The functional relationship between stress relief treatment and tensile strength of a production TG3 R1 pressure tube will be established and the microstructure investigated. A recommendation will be made to NSD on the stress relief treatment for further production tubes.

5.3.2 End-of-Life Deformation Characteristics

A contract will be placed with the CEA to irradiate 16 fracture toughness, 16 delayed hydrogen cracking and 16 tensile specimens in one of the high flux reactors in France at a temperature of 280°C at a fast flux of 2×10^{18} n/m^2/s, E>1 MeV. The target irradiation dose will be 1.5×10^{26} n/m^2, E>1 MeV, with the option to extend the fluence to 3.0×10^{26} n/m^2. Irradiation is expected to start in late 1988.

5.3.3 Dislocation Substructures in Zr-Nb Alloys

TEM studies of dislocation substructure in Zr-Nb alloys will be continued with a view to understanding their role on the mechanical properties and irradiation growth response of the alloy. The main experimental part related to analysis of the α/β interface structure and α/α-low angle boundaries is nearly completed and further effort will be focused on the summary of these results to the point for an external publication.

The microstructure of the Zr-2.5% Nb pressure tube extruded at much lower extrusion ratio (4:1 as opposed to 7.8:1 for the TG3 pressure tubes and 11:1 for the current pressure tubes) will be evaluated.

5.4 Materials Properties for CANDU Reactor Core-Component Alloys

5.4.1 Elevated Temperature Properties for Zr-2.5 Wt% Nb Pressure Tubes

Over the past few years, several test programs were carried out which generated a large data base for the elevated temperature deformation behaviour of Zr-2.5 wt% Nb pressure tube material for temperatures in the range 500-1500°C. This information has been used extensively by reactor safety analysts in the analysis of various hypothetical loss-of-coolant accident scenarios.

More recently, the need for materials data for the Zr-2.5 wt% Nb pressure tubes has shifted to lower temperatures. There is now very keen interest to obtain detailed flow curves for Zr-2.5 wt% Nb PT material over the temperature range 100-400°C. This need for more detailed stress-strain data arises from the increasing use of sophisticated and powerful computer codes in the flaw and materials integrity evaluation of critical engineering components, such as the pressure tubes in CANDU reactors.

To fulfill this apparent need, the appropriate tensile tests will be performed when the requests and/or need arises. In addition, consultation will be provided to other personnel within Ontario Hydro who require advice on the elevated temperature materials properties of Zr-2.5 wt% Nb pressure tubes.

5.5 Materials Properties for Zircaloy-4 Fuel Cladding

A proposal entitled "Steam Oxidation, Biaxial Deformation and Beryllium-Assisted Crack Penetration Behaviour of CANDU Reactor Zircaloy-4 Fuel Cladding at Elevated Temperatures" was submitted to the CANDU Owners Group for financial support in 1988. The main objectives of the proposed test program are:

1. Increase the available data base for the oxidation kinetics of parent and braze HAZ cladding materials to oxidation in a steam environment;

2. Develop a test procedure and carry out preliminary experiments needed to assess fuel cladding failure due to Be-braze assisted cracking at elevated temperatures in steam.

We plan to carry out a test program which will entail oxidizing cladding specimens in a steam environment over the temperature range 600-850°C. To obtain the relevant materials data, selected oxidized specimens will be subjected to the appropriate mechanical testing and metallurgical evaluations. The new data will then be used to modify existing criteria and/or develop new criteria for failure and "handling" of oxidized fuel cladding which have been subjected to degraded cooling conditions.

We also plan to develop a test facility which is capable of heating Zircaloy fuel cladding to temperatures in excess of 1000°C in a steam environment. Once this test facility has been developed, we will perform some preliminary tests at elevated temperatures in a steam environment. We anticipate that the information obtained can be used to establish a test program that is capable of quantifying the role of the Be-braze assisted cracking mechanism to fuel cladding failure at elevated temperatures in steam.

5.6 General Materials Testing

Over the past few years, a very versatile computer-controlled Materials Characterization Test Facility and several experimental test procedures were developed for use in the High Temperature Deformation Laboratory. In 1988, we will endeavour to seek new applications for this very powerful test facility. Possible new areas include metal-matrix composite materials and materials for the next generation of power plants - namely, fusion reactors.

5.7 Materials Properties for CANDU Reactor Ex-Core Component Alloys

5.7.1 Darlington PHT Piping Material

In early 1988, the supplementary tests recommended in OHRD Report No. 87-319-K, to complement the PN-229 weld metal test program will be completed. The need for further tensile testing will be assessed and discussed with the appropriate OHRD and Mechanical Design Department personnel.

6.0 1988 PROGRAMS - FRACTURE MECHANISMS UNIT

6.1 Blisters

6.1.1 Blister Fracture

This program will be continued. The need for the development of the best understanding of blister fracture has been emphasized by the recent measurements of deuterium in Pickering 4 tubes which indicate a higher ingress rate than was thought likely six months ago. The tests remaining from the 1987 program will be carried out and more emphasis will be placed on growing blisters under stress to fracture. The testing method required to do this in an effective way is currently under review. Problems of stress relaxation in the clamped tube which was used to grow blisters under stress in 1987 must be resolved.

It may be necessary to redirect some of the efforts in other hydride related areas or to hire new staff in order to carry out an effective blister fracture program in a short time.

6.1.2 Probabilistic Modelling

This program remains as the key user of the TRAC program results. Refinements of the model will be required to better describe blister growth. However, it is very likely that there will be significant demands for assessments using the model as it currently stands. In order to free up the manpower required for extension of the model, some of the use of the model may be transferred to client groups. Documentation of the current model will be required in the near future for the AECB.

6.2 Delayed Hydride Crack Initiation

6.2.1 Rolled Joint Integrity

The test results from zero clearance rolled joints exposed to simulated service conditions will be examined and compared to results from small crack initiation test specimens. The four rolled joints with machined defects on the ID surfaces near the burnish marks will be monitored

continuously using the acoustic emission technique. Periodic inspection will include an ultrasonic mapping of machined defects in the middle section of the pressure tube. Post test metallography will determine the hydride distribution.

6.2.2 Load Maneuvering

The experimental program will be a continuation of the 1987 program, completion of the scoping test matrix will provide the information to decide if further tests will be necessary. Both thermal and mechanical cyclic loading will be used and metallography coupled with the potential drop method will help to identify the onset of crack initiation.

6.2.3 Crack Initiation in Pressure Tube Materials

In addition to the continuation of 1987 programs, a more fundamental study will be undertaken to improve our understanding of the crack initiation criteria under increasing hydrogen and end-of-life service conditions. A better theoretical understanding will be complementary to the application/development of a model to predict the time to crack initiation.

The current four point bend crack initiation tests will be continued and more data are expected as the test times increase. The ongoing sensitivity analysis to estimate the accuracy of the assumed parameters in the probability of crack initiation model will be completed.

The new project will be a parametric test program using the notched tensile specimen to study the relation between the crack nucleation stress and the following variables:

- hydride morphology at the notch tip
- hydride growth during hold time
- test temperature
- notch geometry
- matrix strength

Specimens with the desired microstructure at the notch tip will be prepared and then tested to crack nucleation to determine the critical load for hydride cracking.

6.2.4 DHC Initiation from Cracked Blister

The ongoing four point bend crack initiation tests to determine if delayed hydride cracking will start from cracked blister will be completed.

6.3 Delayed Hydride Crack Growth Studies

The high deuterium level measured in P3L09 and the possible fast pick-up rate have raised further concerns regarding DHC in pressure tubes. The 1988 program will address the following:

(1) The parameters controlling crack tunneling will be determined using small specimens. The parameters will include material strength, test temperature, applied K and stress state.

(2) Develop a test procedure to simulate the effects of continuous hydrogen ingress. Possible methods are temperature gradient, accelerated corrosion and electro-migration.

(3) The conditions required for cracking to occur at the outlet end temperature will be investigated in terms of thermal history, material strength, stress state and hydrogen concentration.

(4) Completion of the program to study the effects of the K maneuver on DHC behaviour.

6.4 Hydride Morphology in Pressure Tubes

The fracture toughness of pressure tubes can be strongly affected by the orientation of hydrides that are present in the tube. Cirumferential hydrides are much preferred over radial or mixed hydrides in terms of fracture toughness. In this program, we will examine ways of reversing non-circumferential hydrides to the axial-circumferential plane.

6.5 Hydride Precipitation

Further studies on the role of residual grain-interaction stresses on hydride precipitation in Zirconium alloy will be continued. The research is intended to have twofold objective, i) to assess the role of twinning on hydride precipitation in coarse-grained Zircaloy-2 rod, ii) to attempt to assess the role of residual grain-interaction stresses on hydride morphology in fine grained zirconium alloys (Zr-Nb pressure tube or fine-grained zircaloy).

Neutron diffraction technique will be employed to measure internal stresses and optical and electron-metallographic techniques will be used to study hydride morphology.

6.6 EPRI Project RP1890-5: Corrosion Fatigue Boiler Tube Failures in Waterwalls and Economizers

The main objectives of this project are to determine the root cause of corrosion fatigue in thermal plant boiler tubes and to find solutions to the problems. The role of OHRD in this project is to:

1. Develop a discriminatory laboratory test procedure whereby corrosion fatigue cracking can be reliably generated under controlled conditions in a tube specimen;

2. Apply the developed test technique to a program of specific tests to be defined on the basis of inputs from field monitoring (Task 1), stress analysis (Task 2), and the autoclave test portion (Task 3) of the project.

In the first quarter of 1988, we plan to carry out a series of tube specimen tests to thoroughly evaluate the new compression adapters and extension rods for cyclic loading of the boiler tube specimens. Once this has been accomplished, we will then carry out tests to quantify the test conditions needed to produce representative waterside corrosion-fatigue cracks in boiler tube specimens under specified water chemistry, temperature and pressure conditions. The next phase of the test program will entail developing an appropriate method for the detection of crack initiation and/or early crack growth in a representative boiler tube specimen. To accomplish this task, we will evaluate the efficacy of a compliance technique, as well as investigating other NDE techniques.

In the third and fourth quarters of 1988, we plan to carry out the text matrix for the experimental program. The conditions for the above tests will be defined by the results of the field survey and the B&W autoclave small specimen tests. The experimental test matrix is anticipated to consist of about 10 tests.

6.7 **The Nondestructive Evaluation of Hydrogen in Pressure Tube Material**

The development of a prototype tube measurement system will continue into 1988. The main purpose of this system is to investigate the effects of probe contacts and provide a simple method for resistivity measurements of irradiated pressure tubes. The development requires solutions to the following problems:

1. Removable Contact Probes
2. Removable Heating System
3. Deployment System

The present solution to the probe problem is to employ a probe which consists of tungsten carbide bit which is capable of penetrating the oxide coating and provides good electrical contact. The bit was tested on corrosion samples with good results. A removable heater assembly and deployment system is under development. New analysis and control software is also under development.

The first set of tests will remeasure tubes with spot welded potential contacts and thermocouples. Possible sources of noise or spurious signals during measurement will be isolated during these measurements. These tests will also confirm the effectiveness of improved signal conditioning system.

The second set of tests will be a more practical model with retractable contacts and heaters. This rig will be used to study the effects of removable contacts and thick oxides upon resistivity measurements. This rig should allow preliminary resistivity measurements on actual sections of irradiated pressure tubes.

At this point, a research program investigating the effects of irradiation upon the accuracy of the test method will begin. This will entail determining the effects of irradiation upon TSS and the behaviour of hydride dissolution and precipitation.

Calorimetric studies, which are complementary to resistivity studies will also be performed. These measurements will allow a more intensive examination of the processes involved and permit calibration of the resistivity results.

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REPORTS

87-6-K	Assessment of Blister Formation in Pickering 3 and 4 Reactors. D.V. Leemans and M. Leger
87-10-K	Oxidation and Biaxial Deformation Behaviour of CANDU Zircaloy-4 Fuel Cladding in the Temperature Range 873-1123 K E.T.C. Ho and A. Donner
87-36-K	Materials Integrity Section - A Review of 1986 and Proposals for 1987. R.G. Fleck
87-41-K	Calorimetric Study of Hydrogen in Zr-2.5 wt% Nb Preliminary Report. K. Tashiro
87-129-H	DHC Initiation at Fuel Bundle Scratches in Pressure Tubes. G.K. Shek and H. Seahra
87-244-P	Assessment of Blister Formation in the Pressure Tubes in the Pt Lepreau Reactor D.V. Leemans and M. Leger
87-319-K	Tensile Properties of AW and PWHT PN-229 Weld Materials in the Temperature Range 20-300°C Under Constant Applied Strain Rate Conditions. E.T.C. Ho and M.J. McGraw
M87-1-K	Examination of Bruce Unit 2 Pressure Tube Offcuts by Metallography. G.K. Shek
M87-18-K	An Introduction to a Multi-Tasking Data Logger. W.K. Lee
M87-21-K	Proposal to Determine the Effects of Temperature and Pressure Maneuver on DHC Behaviour. G.K. Shek
M87-24-K	Evaluation of Fretted Marks in Pickering Unit 8 Pressure Tubes as Initiators for Delayed Hydride Cracking G.K. Shek
M87-33-K	Hydrogen Diffusion Tests on P8 Fretted Tubes W.K. Lee, M. Leger, O. Kiir
M87-48-K	Trip Report: Evaluation of High Fast Neutron Flux Test Facilities R.A. Holt and R.G. Fleck
M87-49-K	Trip to Chalk River - Apr 2 R.A. Holt

M87-60-H Evaluation of Pilgered Zr-2.5 Nb Pressure Tubes
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M87-61-K DHC Behaviour Under Flow Injection Thermal Condition
G.K. Shek and H. Seahra

M87-73-K Proposal to Evaluate the Steam Oxidation and Biaxial Deformation Behaviour of CANDU Reactor Zircaloy-4 Fuel Cladding Samples in the Temperature Range 673-1123 K
E.T.C. Ho

M87-112-K Hydride Orientation in Zr-2.5 Nb Pressure Tubes
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ME87-1-K Analysis of Inclusions in Pickering Unit 5 Pressure Tube Offcut D122 Front.
G.K. Shek

PUBLICATIONS

Corrosion Fatigue Cracking in Fossil-Fueled Boilers, Journal of Materials Engineering, Vol 9, No 2, p 163, 1987
H.J. Westwood and W.K. Lee

Microstructural Effects on the Irradiation Growth of Zr-2.5 Nb, Zirconium in the Nuclear Industry, STP 939, September 1987.
J.D. Parker, V. Perovic, M. Leger and R.G. Fleck.

Hydride Blister Growth in Zr-2.5 Wt% Nb Pressure Tube Material, Proceedings of Fuel Channel Technology Seminar, February 1987, CRNL 2950, OH D&D GEN 85431.
M. Leger and T.P. Byrne

Creep and Growth Tests of Pressure Tubes, Proceedings of Fuel Channel Technology Seminar, February 1987; CRNL 2950, OH D&D GEN 85431,
J.D. Parker, A.R. Causey and R.G. Fleck

PRESENTATIONS

"Mechanisms of High Fluence Irradiation Growth of Zirconium Alloys" at the International Workshop on Fundamental Mechanisms of Irradiation Creep and Growth, Hecla, Manitoba, June 1987.
R.A. Holt

"Observations of the Bainite/Austenite Interface in Steels". Phase Transformations' 87 - International Conference on Phase Transformations in Solids, University of Cambridge, England
Chengji Li, V. Perovic, G.R. Purdy

"The Effect of Intergranular Stresses on the Texture Dependence of Irradiation Growth of Zirconium Alloys". Presented at the International Workshop on Fundamental Mechanisms of Irradiation Creep and Growth, Hecla, Manitoba, June 1987.
A.R. Causey, C.H. Woo and R.A. Holt

"Effects of Temperature Neutron Fluence on Irradiation Growth of Zr-2.5 wt% Nb". The International Workshop on Fundamental Mechanisms Irradiation Creep and Growth, Hecla, Manitoba, June 1987.
R.G. Fleck, R.A. Holt, V. Perovic and J. Tadros

CONFERENCES AND TECHNICAL MEETINGS

CFMS Workshop on Fracture Mechanics for Fracture Control of Structures. March 3-5, 1987, Toronto, Ontario,
G.K. Shek

Visit to CEA Saclay, France to Discuss Irradiation Test Facilities, March 9-10, 1987,
R.G. Fleck, R.A. Holt.

Visit to CEA Grenoble to Discuss Irradiation Test Facilities, March 11-12, 1987
R.G. Fleck, R.A. Holt.

Visit to AERE Harwell, U.K. to Discuss Irradiation Growth Program in Dido, March 16-17, 1987
R.G. Fleck, R.A. Holt

Engineering Applications Using Lotus 1,2,3,. Course offered by Binary Engineering, March 19-20, 1987.
W.K. Lee, G.K. Shek, H. Seahra

Meeting at CRNL to Discuss Fast Flux Assessment in High Influence Irradiation Facilities, April 2, 1987
R.A. Holt

Meeting at CRNL to Discuss Calandria Tube Design Equations, May 2, 1987
R.A. Holt

Meeting at AECL-SP to Discuss Calandria Tube Serviceability, May 26, 1987
J. Tadros

Attend International Workshop on Fundamental Mechanisms of Irradiation Creep and Growth, Hecla Island, Manitoba, June 22-25, 1987
R.G. Fleck, R.A. Holt, V. Perovic

Attend Third International Conference on Fatigue and Fatigue Thresholds, University of Virginia, Charlottesville, Virginia, June 28-July 3, 1987
E.T.C. Ho

International Conference on Phase Transformation in Solids, University of Cambridge, July 6-10, 1987
V. Perovic

Visit to U.K. to the AERE, Harwell, U.K., July 13-14, 1987
V. Perovic

Conference of Metallurgists, Winnipeg, Manitoba, August 22-26, 1987
W.K. Lee

Meeting with S.R. MacEwen at CRNL to Discuss the program on Hydride Precipitation, August 27, 1987
V. Perovic

Meeting with Prof. Smith of University of Manchester at WNRE, Pinawa, Manitoba, September 8-11, 1987
W.K. Lee

Meeting with B.A. Cheadle, A.R. Causey, V. Fidleris at Chalk River to Discuss COG Programs, September 14-15, 1987
R.A. Holt,

Meeting with B.A. Cheadle at Chalk River to Discuss COG Programs, September 14, 1987
R.G. Fleck

Meeting with AECL-RC and AECB at Chalk River, September 15-16, 1987
R.G. Fleck

Canadian High Technology Week '87, Toronto, Ontario, September 28, 1987
G.K. Shek

Meeting with A. Rogerson, Risely Nuclear Labs (UKAEA) to discuss irradiation growth, October 26, 1987
R.A. Holt

Meeting with B. Edwards, B. Shelden, R. McElroy, R. Ainsley, AERE Harwell (UKAEA) to discuss irradiation growth experiments in Dido, October 30-November 2, 1987
R.A. Holt

Meeting J.M. Cerles, CEN de Saclay (CEA) to discuss Creep Growth and Fracture Experiments in SILOE and OSIRIS, November 2-3, 1987
R.A. Holt

Meeting with M. Masson, CEN de Grenoble (CEA) to discuss Creep and Growth Experiments in SILOE, November 4, 1987
R.A. Holt

Attend BNES Conference on Irradiation Effects in Nuclear Core Components, Bristol, England, October 27-29, 1987
R.A. Holt